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REACTOR TECHNOLOGY

REVISIONS TO EXPERIMENTAL GAS COOLED REACTOR FINAL HAZARDS SUMMARY REPORT

Volume I. Description and Hazards Evaluation

April 1965

Oak Ridge Operations Office, AEC
Oak Ridge, Tennessee

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REVISIONS TO
EXPERIMENTAL GAS COOLED REACTOR
FINAL HAZARDS SUMMARY REPORT
OCTOBER 10, 1962

VOLUME I
DESCRIPTION & HAZARDS EVALUATION

The revised pages and figures listed below are intended to replace those issued in the original publication under date of October 10, 1962. These revised pages have been punched so that they may be inserted in those copies which were issued in loose leaf format. Holders of the bound editions may desire to disassemble their copies, punch the pages, and reassemble in loose leaf format. After insertion of these replacement pages, the original cover sheet should be marked to indicate the inclusion of revised material.

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OAK RIDGE OPERATIONS OFFICE

UNITED STATES ATOMIC ENERGY COMMISSION

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1. INTRODUCTION

In April 1959, the United States Atomic Energy Commission initiated, through its Oak Ridge Operations Office, the design and construction of the Experimental Gas Cooled Reactor (EGCR). The responsibility for the design of the facility has been assigned to Kaiser Engineers with Allis-Chalmers Manufacturing Company as nuclear subcontractor. The Oak Ridge National Laboratory was assigned the responsibility for the detailed design of the reactor fuel and the reactor control rods. Union Carbide Nuclear Company was assigned the responsibility for the detailed design of the emergency cooling system and the procurement of the equipment for the emergency cooling system, the reactor fuel, and the reactor control rods. The Laboratory also served as technical advisor to the AEC in a continuing review of the detailed design. The H. K. Ferguson Company was selected as construction contractor, and the Tennessee Valley Authority was selected to operate the facility for the Atomic Energy Commission. The design of the plant was based upon previous design studies of gas cooled reactors by Allis-Chalmers, Kaiser Engineers, and the Oak Ridge National Laboratory.

Based upon preliminary hazards analyses reported in ORO-196, construction of the EGCR was initiated in August 1959. Construction is scheduled for completion in the third quarter 1965.

The EGCR is designed with sufficient flexibility that it may be operated solely as a power generating facility or as a combined power generating and experimental facility. Provision is made for the future addition of experimental loops. The plant will be operated without loops at least until 1966. At that time, if experimental loop facilities are to be added to the plant, a supplemental Hazards Summary Report which considers the experimental loop facilities will be prepared.

This Hazards Summary Report consists of two volumes. Volume I contains a detailed plant description and the results of analysis of credible accidents and the associated hazards. Volume II describes those programs established by the operator to assure the safe operation of the reactor. Included is a description of: the organization and training; the testing program prior to and during reactor start-up; and those programs associated with normal and emergency operations, including routine plant tests and inspections, maintenance, decontamination, waste disposal and fuel handling. Radiation standards governing all operations are defined, and the radiation control measures required to meet these standards are described.

A summary of important EGCR design data (without experimental facilities) is given in Table I.

TABLE I

Summary of Important Design Data

Reactor thermal power	84,300 kw
Blower heat of compression	4000 kw

TABLE I (continued)

Power to steam generators	88,300 kw
Gross electric power	29,500 kw
Plant auxiliaries - power requirement	7600 kw
Plant efficiency	
Gross	34.7%
Net	25.9%
Reactor coolant	helium
Reactor core coolant flow (not including annuli)	395,000 lb/hr
Reactor core coolant flow (including annuli)	427,000 lb/hr
Total coolant flow	430,000 lb/hr
Coolant charge	1350 lb
Fuel channel outlet gas temperature (at rated power)	1075 F
Fuel channel inlet gas temperature (at rated power)	510 F
Reactor outlet gas temperature (at rated power)	1043 F
Reactor inlet gas pressure (at rated power)	313.7 psia
Reactor outlet gas pressure (at rated power)	303.4 psia
Coolant system pressure drop (at rated power)	16.75 psi
Steam flow (100% load)	256,000 lb/hr
Pressure at throttle	1250 psig
Temperature at throttle	900 F
Fuel loading (U in UO ₂)	12.23 metric tons
Fuel enrichment (initial core)	2.46% U ²³⁵
Fuel enrichment (equilibrium core)	2.25% U ²³⁵ in U ²³⁸

TABLE I (continued)

Reactivity requirements - (k_{eff}) - initial core

Operating temperature (equilibrium Xe and Sm)	1.087
Operating temperature (clean)	1.115
Cold, clean	1.161
Cold, clean with all rods in	0.920

Fuel element heat flux

Maximum	174,000 Btu/hr·ft ²
Average	64,400 Btu/hr·ft ²

Neutron flux (thermal)

Maximum	1.92×10^{13} n/cm ² ·sec
Average	0.78×10^{13} n/cm ² ·sec

Neutron flux (non-thermal)

Maximum	4.29×10^{13} n/cm ² ·sec
Average	1.63×10^{13} n/cm ² ·sec

Maximum-average flux ratio

Axial	1.8 (1.7 - 2.0)
Radial	1.35 (1.2 - 1.35)

Control rods (25 positions available) 21

Absorber boron carbide

Cladding type 304 stainless steel

exceeding design temperature limitations. After three hours it might be necessary either to provide forced cooling or to depressurize partially to prevent the reactor vessel top nozzles from exceeding design conditions. Should the service machine be connected to a top nozzle it might be necessary to have some forced cooling after one hour to prevent overheating of that nozzle.

2.4 Instrumentation and Automatic Features

2.4.1 General

The EGCR instrumentation consists of three distinct systems which are designed to perform specific functions.

a. Reactor Safety System

Function: To shut down the reactor in any situation which is potentially hazardous to the plant, operating personnel, or the public.

b. Plant Control System

Function: To control the reactor and steam systems during normal operation.

c. Indications and Alarms

Function: To indicate and record plant operating parameters and to direct the attention of the reactor operator to abnormal conditions.

The reactor safety system and the plant control systems are completely independent. Each system includes its own detectors and circuitry; neither system is dependent upon the other for proper operation, nor does failure of any portion of one system affect the operation of the other system in any way.

2.4.2 Reactor Safety System

The reactor safety system which includes those instruments and associated devices measuring critical reactor and plant parameters, generates signals and takes automatic actions to avoid abnormal conditions which are unsafe or potentially unsafe. The system utilizes coincident circuitry to minimize the possibility of unnecessary scrams due to a faulty safety system channel.

Transistors are employed throughout the safety system to assure reliability, speed, and sensitivity. Except for neutron flux in the start-up range (two channels) and loss of power, a minimum of three channels are installed for each measured variable. The output from each channel is fed to each of three mixer stages. The output of each mixer stage is fed to three power stages which actuate the scram relays. Thus, to initiate a reactor scram, a minimum of two trip signals associated with a specific instrument channel must be applied to the mixer stages. A minimum of two trip signals from the mixer stage output must be applied to the power stages. Two out of three power stages must trip to initiate scram action.

The safety system utilizes components of proven reliability and each component is arranged to fail-safe. Fail-safe features and reliable instruments and system components provide the EGCR with a safety system which will adequately protect the reactor against improper operation. The safety system does not require any assistance from operating personnel to perform its basic function of shutting down the reactor. The redundancy of coincident circuitry allows frequent testing of each individual safety channel, and, when necessary, maintenance can be accomplished without affecting reactor operation or reducing the effectiveness of the protection supplied by the safety system.

2.4.3 Plant Control System

The control system includes those instruments and devices which measure plant variables and act to maintain the measured parameters within prescribed limits. The reactor control system, like the reactor safety system, utilizes reliable instruments and components. Two reactor parameters control the reactor power and temperature within the system. Inlet coolant temperature controls reactor power; outlet coolant temperature controls the speed of the reactor coolant blowers which regulate the temperature within the system. A deviation from the inlet temperature set point produces an error signal to a controller which alters control rod positions and a deviation from outlet coolant temperature set point varies reactor coolant blowers speed to bring the reactor outlet coolant temperature to the set point. Both inlet and outlet coolant temperature set points are reset by reactor power level.

The reactor control system uses multiple channels with auctioneering circuits to operate from those signals which indicate the highest temperatures. This auctioneering prevents reactor operation above design power or in excess of design temperatures in the event of an instrument failure.

2.4.4 Indication and Alarms

The remainder of the instrumentation, which only measures a parameter or performs process control or minor local control, is considered plant process instrumentation. Alarms are installed to alert the operator of any approach to an off-design condition.

2.5 Specific Safety Features

The inherent safety of the EGCR, combined with limited rate of reactivity insertion by control rods and the completely independent and reliable safety system, afford adequate protection against nuclear excursions. A number of design features have been incorporated to assure plant safety.

The maximum rate of reactivity insertion due to manipulation of the control rods, has been carefully selected so that reactor components cannot be damaged, even in the event of uncontrolled rod withdrawal. Short reactor periods are precluded. Analogue computer simulations for rod runaway accidents in the source, intermediate, and power ranges indicate that the shortest reactor period that could occur is approximately 0.6 sec.

The reactor safety system generates signals to shut down the reactor in the event of operator error, malfunction of the control system, or equipment failures which would cause the plant to approach an unsafe condition or to exceed its design limitations. The safety system is designed for extreme reliability by the use of multiple channels measuring the same parameter and the use of reliable components. Output signals from these multiple channels are passed through coincidence circuitry so that at least two signals which indicate need for scram action are required to scram the reactor. Malfunction of one channel will fail safe, i.e., will be in the direction of a scram.

One of the more important safety features is that of containment of radioactivity. There are three containment barriers to the release of radioactivity; the fuel element cladding, reactor coolant system and the containment shell.

The fuel element cladding material is type 304 stainless steel for which extensive strength data up to 1800 F are available. A statistical analysis has shown that during normal operation, the probability is negligibly small that the fuel cladding temperature of any fuel element in the reactor will exceed 1752 F. Further, the fuel programming is such that the internal fission

vicinity of the point of entry into the soil. It is concluded that such ground-water flow as may exist below the surface in the soil surrounding the site will be small and slow (a few feet per week) and that natural chemical fixation will reduce the level of the activity of mixed nonvolatile fission products more than 90%.

The Conasauga shale formation is an extensive formation which is quite heterogeneous in structure and relatively low in permeability.⁽⁹⁾ The depth and extensiveness of the formation provide a large capacity for decontamination of any contaminated liquid reaching it, and the slow rate of percolation improves the efficiency of decontamination by ion exchange and radioactive decay. However, the heterogeneity of the formation makes difficult the prediction of patterns and rates of the movement of water and hence of the fission products contained therein. It has been noted, however, that most of the seepage is along bedding planes parallel to the strike.

There is very little probability of ground-water contamination in areas removed from the ORNL site, and there are no developed ground-water resources in the area. The lack of ground-water resources is due to the fact that formations present are, for the most part, too impermeable to hold or transmit any significant quantity of water.

The general site geology has recently been supplemented by forty-five core drillings on the site. Analyses of these cores ⁽¹⁰⁾ have verified the presence of the Conasauga formation beneath the site. The base of the Conasauga is a dark red or maroon silty shale with numerous thin beds of laminae of fine-to-medium-grained, light-gray crystalline limestone, with no indication of solution channels. Based on the cores from the site, it was concluded that foundation-bearing loads, 15 to 20 ft below the surface, may be 6 tons/ft². In general, the overburden consists of badly weathered or rotten shale overlain in places by ancient alluvial deposits. The foundation loading of 6 tons/ft² is for relatively fresh and unweathered shale.

3.6 Seismology

In both the design of existing facilities at ORNL and the design of dams and power houses by TVA in East Tennessee, no allowance for earthquake forces has been made. Oak Ridge has been classified by the U. S. Coast and Geodetic Survey as subject to earthquakes of intensity mm-6 measured on the modified Mercalli Intensity Scale.

Information on the frequency and severity of earthquakes in East Tennessee has been obtained both from Lynch⁽¹¹⁾ of the Fordham University Physics Department, and from Moneymaker⁽¹²⁾ of the Tennessee Valley Authority. Both sources indicated that such shocks as occasionally occur in the region are quite common over the world and indicate a very low level of seismic activity. Consequently earthquakes, although considered in the design, will not affect operation of the EGCR.

The TVA records show that the Appalachian Valley from Chattanooga to Virginia has an average of only one or two earthquakes a year. The maximum intensity of any of these shocks is six on the Modified Mercalli (by Wood-Neumann) scale. A quake of this magnitude was experienced in the Oak Ridge area on September 7, 1956, and was barely noticeable by either ambulatory or stationary individuals. Structures were completely unaffected. For any one location such as Oak Ridge, the expectancy of an earthquake of this type should be one in every few years.

The Fordham University records indicate an even lower quake frequency; however, the recorded severity of the observed quakes is the same. Lynch further concluded "that it is highly improbable that a major shock will be felt in the

area (Tennessee) for several thousand years to come."

A design seismic loading of 0.05 g lateral and 0.025 g vertical, based on mm-6 intensity earthquake, was set as a result of a conference⁽¹³⁾ held July 2, 1959, attended by representatives from ORO (AEC), the U. S. Coast and Geodetic Survey, Kaiser Engineers, and ORNL, and subsequent correspondence. The selected value is greater than that of any recorded shocks and is intended to cover any that might reasonably be anticipated in the foreseeable future. Other meetings held with Dr. Housner, California Institute of Technology, verified the adequacy of the selected seismic design basis. A complete seismic analysis has been made of the reactor pressure vessel and core.⁽¹⁴⁾

3.7 Population Distribution

The total population of the four counties (Anderson, Knox, Loudon, and Roane) closest to the site is 370,145 of which 177,255 are concentrated in cities with populations greater than 2500 persons. The rural population density in these four counties is about 135 persons/mi². Table 3.7.1 lists the 1960 U. S. Census populations in the 14 counties nearest the site, with a total population of 589,639. Knox County accounts for 247,974, or 42% of this total. From Figures 3.1.1 and 3.3 the significant fact is evident that no population concentrations lie in the cones of the most frequent wind directions.

In the area controlled by the AEC, the X-10, K-25, and Y-12 plant sites are approximately 2 mi west, 6.5 mi west, and 3.5 mi north of the EGCR site respectively. The closest concentrated residential area is the Gamble Valley area of Oak Ridge, 4.1 mi from the EGCR site, while the population center of the city of Oak Ridge is approximately 5.5 mi north of the site. The edge and center of the city of Knoxville are about 15.5 and 18 mi east of the site respectively. The population in the uncontrolled areas across the Clinch River from the site is small, as shown in Table 3.7.2.

nitrogen. All pellets are visually inspected to assure uniformity. One pellet of each hundred is measured and weighed to determine the bulk density. An additional one-percent sample is heated and weighed to confirm the dryness. The stack height and stack weight of each column of pellets is recorded.

The end caps and spacers are machined from stainless steel bar stock and are subjected to dye-penetrant, visual and dimensional inspection. The spiders are precision cast and machined to the final dimensions. The spiders are inspected by radiography, dye-penetrant and visual means.

The fabrication of the fuel elements has been demonstrated to be straightforward.⁽¹⁾ A mid-plane spacer is brazed to each tube and the bottom end cap is welded in place. The stack of 36 fuel pellets, plus a magnesium oxide spacer at each end, is then loaded into the tube. The tube is placed in a helium-filled chamber where the top end cap is inserted and welded. Inspection consists primarily of dye-penetrant and radiographic examination and helium leak tests of each weld. Fuel elements showing leaks in excess of 7×10^{-9} standard cm^3/sec of helium are rejected.

Completion of the fuel assembly consists of orienting the fuel element spacers, welding the fuel elements to the top spider, inserting the fuel elements into the graphite sleeve, and welding the fuel elements to the bottom spider. The shear pins are then inserted and welded to the top spider. Inspection of the fuel assembly consists primarily of passing a gage between the fuel elements and between the elements and the sleeve to assure minimum spacing. The fabrication records on the fuel assemblies and all component parts will be made available to the reactor operator for use in fuel programming and fuel failure analysis.

4.2.4 Support in Reactor

Dummy assemblies at the top and bottom of each fuel channel serve to position the column of six fuel assemblies within the active region of the reactor core. Figure 4.2.4.1 shows the design of the bottom dummy assembly. The weight of the fuel column is transmitted to the fuel channel insert through three retractable support legs in the bottom dummy assembly. The fuel channel bottom insert is supported by 1/8-in. shoulder in the core bottom plate.

The support legs are sized so that the total weight of the fuel column can be supported by one leg.

The bottom dummy assembly is unlatched from the fuel channel insert by the charge machine fuel ram. The ram is raised until the plunger in the ram contacts the drive shaft in the bottom dummy. Initial movement of the ram after contact unlocks the clutch on the bottom dummy adjustable orifice. Further movement of the ram retracts the moderator seal and, after lifting the dummy and fuel column, retracts the support legs. The preceding movements are controlled through incremental jogging of the fuel ram drive motor. Additional upward movement raises the fuel column and top dummy assembly to a point at which the top dummy ball lock restrains the column from further upward movement. A weight reading gives a positive indication of this fact to the charge machine operator.

When the fuel column, dummies and ram head are at their uppermost limit of travel, the operator may either withdraw the fuel by lowering the ram head; or install the fuel column by retracting the central plunger to extend the support legs, and lower the column until supported by the fuel channel insert. With the weight of the dummies and the fuel column supported by the support legs, the continued withdrawal of the ram head permits actuation of the moderator seal.

When the bottom dummy is in position in a fuel channel, the support legs are

CORE BOTTOM DUMMY ASSEMBLY
FIG. 4.2.4.1

CORE FUEL CHANNEL EXTENSION

FIG 4 2 4 2

held in the extended position by the weight of the dummies and the fuel column, and by springs.

The top dummy assembly (Figure 4.2.4.3) is installed in the top of the core fuel channels by the service machine. When the fuel channel is empty, the top dummy assembly is supported by a 1/16-in. shoulder on the assembly which rests on the steel insert in the top of each fuel channel (Figure 4.2.4.2). When the fuel channel is loaded, the top dummy assembly is supported on the top fuel assembly.

Ejection of the top dummy assembly from the channel is prevented by a ball lock assembly. The ball lock is unlatched by the service machine fuel grapple with a vertical upward motion. The surface on the latch actuator just behind the locking balls, when the latch actuator is in its locked position, is slightly recessed. This recess transforms any upward force on the top dummy assembly into a locking force on the latch actuator.

4.2.5 Fuel Channel Orifices

The coolant flow to core fuel channels is individually regulated by means of an adjustable orifice in each of the bottom dummy assemblies so that the coolant gas temperature leaving each fuel channel is 1075 F at full reactor power. The orifices are adjusted prior to installation of fuel into the core and also after installation of the fuel within a fuel channel. The adjustment is performed by the charge machine. Coolant temperatures are measured by a thermocouple at the exit of each fuel channel. Coolant flow within each fuel channel is divided between the inside of the fuel assembly sleeves and the annulus formed by the outside of the sleeves and the inside of the fuel channel. This division of flow is accomplished by means of a fixed orifice in the bottom dummy and a narrow annular flow path formed by the top dummy and the fuel channel insert. These restrictions are so sized that the temperature of any exposed graphite will not exceed 1100 F at full reactor power.

The adjustable orifice shown in Figure 4.2.4.1 regulates coolant flow by the positioning of the tapered plug relative to a fixed orifice. Female threads in the plug engage male threads on a hollow shaft located within the plug. The hollow shaft is permanently attached to the body of the dummy. The plug is locked to the hollow shaft by a positive clutch and spring action, except during adjustment. The plug is raised or lowered by the charge machine fuel ram which engages the drive shaft, raises the shaft to disengage the clutch, and then rotates the shaft to position the plug. The bottom dummy assembly remains locked in the fuel channel insert during orifice adjustment.

The tapered surface of the plug is designed to maintain a linear relationship between number of rotations of the shaft and the coolant flow through the fuel channel over most of the controlled flow range. The largest deviation from linearity occurs near the wide-open position where more rotations are required for a given change in coolant flow.

Mechanical stops limit the minimum orifice-plug flow area so that cladding temperatures will be limited in the event of excessive closure in the bottom dummy adjustable orifice. The ratio of the channel power to the average power ranges from 0.5 at the edge of the core to 1.35 near the center of the core. For purposes of placing the mechanical stops, the core has been divided into three zones based on the fuel channel power ratio as follows:

<u>Zone</u>	<u>Mechanical Stop Length (Inches)</u>	<u>Channel Power Ratio</u>	<u>Flow Range (Pounds per Hour)</u>
1	1.15	0.50 to 1.35	900 to 2400
2	2.15	0.73 to 1.35	1300 to 2400
3	3.40	1.00 to 1.35	1800 to 2400

FIG 4.2 4.3 TOP DUMMY ASSEMBLY

The flow to the moderator coolant annulus is supplied through fixed orifices in the bottom dummy, downstream of the adjustable orifice. Flow in this annulus is also restricted at the top dummy assembly by means of a long, narrow, annular labyrinth formed by 21 circumferential grooves on the outer surface of the top dummy and a smooth inner surface of the core fuel channel insert. The orifices and restriction are so sized that for a fuel channel operating at the maximum anticipated fuel channel power, the coolant flow in the annulus will be about 135 lb/hr. Under these conditions, the pressure drop across the bottom dummy fixed orifice is about 50% of the total annulus pressure drop and the pressure drop across the top dummy orifice is about 50%. For the lower power fuel channels, moderator coolant flow is reduced proportionally to the fuel column coolant flow. Thus, the distribution of coolant between fuel and moderator remains approximately constant over the full range of adjustable coolant flow. ORNL is running tests to confirm the distribution of coolant between the fuel and the moderator over the full range of adjustable flow.

4.3 Reactor Core Structure

The core of the EGCR consists of 120 structurally independent graphite columns supported upon and keyed to a 3-in. thick plate which is bolted to the core support grid. The columns are positioned at the top by a restraint lattice which maintains the vertical alignment of the graphite columns. Fuel channels are bored into those graphite columns which form the moderator section of the core. A gas seal, graphite-to-plate, installed at the interface of the graphite column and the support plate at each fuel channel, minimizes coolant leakage from the fuel channels to the gaps between the graphite columns. The graphite structure is peripherally restrained by a restraint structure located at the core mid-plane. Figure 4.3 shows a cutaway model of the entire core structure.

4.3.1 Design Criteria

The primary functions of the core structure are:

- a. To provide the structural framework to maintain the nuclear fuel in a critical configuration and to position control rods
- b. To slow the fast neutrons to thermal energies and maintain the fission chain reaction
- c. To permit efficient removal of fission heat by the reactor coolant.

The design requirements of the core structure are:

- a. To fulfill the above three functions under all conditions of operation
- b. To provide for detection of faulty fuel elements and the removal and replacement of any or all fuel
- c. To permit insertion of control rods for all credible operating conditions.
- d. An operating life of 20 years (All core components which are not readily replaceable must be designed for operation with a core fuel channel outlet coolant temperature of 1200 F, coincident with the equivalent of 10,000 thermal cycles between temperatures associated with 20% and 100% power operation.)
- e. To withstand a lateral ground acceleration of 0.05 g and a vertical ground acceleration of 0.025 g.

radiation-induced shrinkage of graphite upon the life of the EGCR core.(3) Although the best available data were used in the calculations, the necessity to extrapolate beyond that data render the conclusions of the study speculative and subject to experimental confirmation. In the course of the study of the radiation shrinkage effects, it was necessary to establish values for the mechanical properties. As with the shrinkage data, it was necessary to extrapolate beyond existing data for stress-strain relationships, creep rates, and effects of temperature and irradiation on the properties of the graphite. Because of the complexity of the graphite columns and of the neutron flux within the columns, it was necessary to set up a simplified analytical model and to make some simplifying assumptions. The model was established by dividing the 16 in. square graphite column into four 8 in. square segments, each containing a fuel channel. The assumption was then made that this cell would behave in much the same manner as a hollow, thick-walled cylinder.

The irradiation shrinkage-induced stresses are of two distinct types:

1. Internal stresses caused by the variation of the fast neutron flux along the radius of each unit cell consisting of a fuel channel and surrounding graphite moderator
2. Stress caused by restraint of bowing which is related to the over-all fast-neutron flux gradient between opposite faces of a graphite column.

The variation of the fast neutron flux along the radius of each cylindrical fuel cell causes differential shrinkage between the fuel channel surface and any other point within the graphite. The resultant stress pattern along any radius of a fuel channel is similar to the stress pattern of a beam in flexure. The presence of from two to four fuel channels in the columns complicates the problems of calculating stresses. However, the simple cylindrical model is considered adequate to describe stresses around any one channel. In those portions of the column cross-section where geometrical fit between model and actual column is not exact, or where the models assumed for adjacent channels overlap, the shape of the neutron-flux gradient gives rise to compressive stress. Since graphite is known to be much stronger in compression than in tension and since the flux gradients of adjacent channels do not greatly differ, it is concluded that the stresses computed for the model of one channel, when considered separately from the remainder of the column, will not differ greatly from the stresses which result from the more complicated mathematical treatment of the entire column cross-section. Similarly, the over-all stress distribution will not be greatly affected by recombining the various models which have been separately considered.

The second major stress problem caused by a differential shrinkage of graphite is that of bowing of individual graphite columns. Bowing results from the over-all flux gradient from face to face of any column. Since the inward faces (those nearest the center of the reactor core) are subjected to a higher flux, their fibers shrink more than those of the outward faces. The effort of the column to accommodate these relative shrinkages results in bowing. The bow is generally outward, because the shape of the flux gradient tends to shorten on the inner face more than the outer face.

If every column were free to bow fully, complete accommodation of all shrinkages would occur. Only minor stresses would be generated internally in a column because of non-linearity of the over-all gradient. However, the closely fitting peripheral restraint bands which encircle

the core limit the extent of column bow, but the accumulation of differential fiber shrinkage continues with continued irradiation exposure. The "restrained bow" is equivalent to an inward deflection of the column from its fully-bowed state. If the columns are considered to behave elastically, the imposed elastic stress can be calculated by the standard stress and deflection formulae.

In 68 of the 120 columns, stresses are not significant. The remaining 52 columns are those in the roughly annular zone centered on the moderator-reflector interface. On the basis of the elastic model analysis, it was found that in certain of the 52 columns noted above, stresses would reach the maximum fiber rupture stress of graphite in flexure, namely, 1800 psi. Based on this maximum stress criterion, some columns were found to reach a limiting stress in about 7 years (shrinkage rates of .02%/1000 Mwd/AT parallel and .01%/1000 Mwd/AT perpendicular).

c. Relief of Stresses

Since graphite is known to behave plastically to some extent, further investigations were made to estimate the effects of creep. Information in published literature and results from other laboratory tests indicate that the stress in the graphite will build up to about 1000 psi and level off at this value.⁽⁴⁾ Subsequently, the local effects of radiation shrinkage and restraint of bowing would be relieved by plastic elongation or creep. Maximum strain, not maximum stress appears to be the limiting factor.

If creep occurs and if stress levels are below the limiting value, the time to failure of the column, subjected to the severest flux gradients, will be approximately 13 years, based on a limiting strain criterion. Column life may be longer because both creep rates and strength are believed to be enhanced by irradiation.

4.3.5 Inspection

During the annual scheduled shutdowns, selected channels in the reactor core can be inspected with the reactor in a depressurized condition.

Borescopes, gages, and other optical tools inserted through the reactor vessel top nozzles will be used to assess dimensional changes in the core structure.⁽²⁶⁾

4.3.6 Research and Development

A continuing investigation is being carried on by the USAEC at Oak Ridge and at Hanford to evaluate the properties and behavior of graphite under conditions of service in the EGCR and in future gas-cooled reactors of EGCR type.

4.4 Control Rods and Drives

4.4.1 General Description

Each control rod is supported from a cable which is wound on a cable drum in the control rod drive mechanism. The general arrangement of the drive is shown in → Figure 4.4.1.1. (prior to addition of an oil reservoir and deletion of the motor → brake). The drive mechanisms are installed in the reactor vessel control rod nozzles, and are supported from a ledge within the nozzles. During installation and removal, the support is transferred to the breech closure. The drives can be individually removed and replaced by the service machine

EGCR Control Rod Drive

FIG. 4.4.1.1

while the reactor is operating at full power.

Each drive mechanism is packaged within a circular steel housing which fits within a reactor vessel control rod nozzle. The lower end of the housing is filled with high-density concrete which serves as a neutron shield. Additional shielding surrounding the cable drum, minimizes gamma radiation levels at the top of the nozzles when the irradiated cable is wound upon the cable drum.

The control rod drive is cooled by a flow of 125 F helium through the mechanism. The gas enters near the top of the control rod nozzle, flows downward into the control rod shroud, into the open center of the control rod, and is finally exhausted from the lower end of the rod to gaps between the graphite moderator columns or to the reactor vessel upper plenum.

The rapid insertion of the control rod, which is necessary during scram, is accomplished by release of a clutch which allows the rod to fall under approximately 0.6 g. The scram clutch is released by interruption of d-c power to the clutch. When the rod reaches approximately the core midplane, a scram control mechanism which is permanently geared to the cable drum decelerates the control rod over its remaining travel into the core.

A removable shroud is installed with each control rod. The shroud fulfills the following functions:

- a. Prevents control rod flutter which could result from turbulent gas conditions in the upper plenum
- b. Contains and directs helium coolant into the control rod
- c. Supports the control rod shock absorber which absorbs the energy of a rod which falls as a result of a cable failure.

The shock absorber and shroud arrangement is shown in Figures 4.4.6.1 and 4.4.6.2. The shock absorber assembly consists of an inconel collet and a control rod adapter attached to the top of each control rod. Normally, the shock absorber is not engaged since the mechanical crash stop in the drive mechanism limits rod travel into the core. In the event of a control rod drive cable failure, the adapter engages the fingers of the collet to dissipate the energy of a falling control rod. The shock absorber is designed to stop the control rod without exceeding the design deceleration forces of the control rod proper.

The control rod design for the EGCR is shown in Figure 4.4.1.2. The rod is approximately 20 ft long, 3-1/4 in. OD, and has an absorber length of about 15 ft. The rod is hung from the drive mechanism by a 3/16-in. stainless steel wire cable and is vertically positioned within a 4-in. diameter control rod channel in the core graphite. In the absorber section, annular rings of hot-pressed B₄C, each 3 in. high are encased in type 304 stainless steel tubing. The tubing is vented through a metallic filter so that no pressure differential will exist across the tubing walls. The control rod is made in segments and is therefore sufficiently flexible to insure entry of the rod into the core even if the channel is not straight. The segments are supported by a stainless steel support rod which occupies a central position within the segments. Annular ball-and-socket joints between segments permit the segments to rotate relative to each other. The segments are kept in contact by a top compression
→ spring to minimize coolant out-leakage at these joints.

Spacers are welded to the inner tube at the mid-point and at the bottom of each

ITEM NO.	QTY.	NAME
2	1	UPPER SPRING SUPPORT
3	1	LOWER SPRING SUPPORT
4	3	TOP CLOSURE
5	1	BOTTOM GUIDE
6	1	SPRING
7	1	LOWER TUBE
8	1	UPPER TUBE
9	3	OUTER TUBE
10	57	BUSHING, 9 & C
11	2	INNER TUBE TYPE I
12	1	INNER TUBE TYPE II
13	2	BOTTOM CLOSURE TYPE II
14	1	BOTTOM CLOSURE TYPE III
15	1	CENTRAL ROD
16	10	SPACER SEGMENT TYPE II
17	2	SPACER SEGMENT TYPE I
18	1	SPIDER
19	4	BOTTOM GUIDE ARM
20	3	FILTER
21	1	BOTTOM CLOSURE TYPE I

GENERAL ARRANGEMENT
DETAIL & GENERAL ARRANGEMENT
SEGMENTED CONTROL ROD

FIG. 4.4.1.2

segment to assure a 5-ft free length of the support rod which will readily bend to give the desired degree of control rod flexibility.

Other features of the design are intended to minimize reactions between movable components and to facilitate insertion into the reactor core. To ensure that this flexibility is maintained throughout the 20 year lifetime, one face of each of the ball-and-socket joints is coated with Hastelloy C weld deposit to prevent self-welding at these joints. All stainless steel surfaces in contact with the B₄C are copper plated to prevent any reaction between boron and stainless steel at elevated temperatures. The bottom end of the rod is fitted with a tapered nose piece to prevent the possibility of a rod hang-up during insertion.

4.4.2 Materials

The materials specified for the components of the control rod and the control rod drive are all in accordance with applicable ASTM, NEMA or AISI specifications. Materials testing to assure quality control is required. The following materials are specified:

a. Control Rod Drive Housing - Carbon Steel, ASTM A-212 Grade B

b. Shielding

1. Neutron

High density concrete with ferrophosphorous aggregate placed by the grouted-aggregate concrete method.

2. Gamma

Cast from melted pig lead conforming to ASTM B-29, chemical or acid lead. No reclaimed or refined secondary lead is permitted. The lead is examined for voids by gamma graphing or gamma probing methods.

c. Cable Drum - Carbon Steel, AISI-4140

d. Gear and Bearing Assemblies

1. Gears

All gears are either AISI-440-C or Nitralloy 135. Gear engagement is made by alternating AISI-440-C with a Nitralloy 135 gear so that there is a dissimilarity of metals between the rubbing surfaces. The gears are hardened by malcomizing or nitriding to obtain required hardness for long life and trouble-free operation.

All spur and bevel gears are 20 degree pressure angle to AGMA Standards and all gears are ground to a surface finish of 32 rms or better. Where bevel and internal gears cannot be ground, they are machined and vapor or wet blasted to a finish of 32 rms or better.

2. Bearings

All ball and roller bearings are fabricated of AISI-440-C and M-2

to move laterally.

The control rod drive mechanism is housed in the upper portion of the control rod drive plug assembly. This assembly is shown on Figures 4.4.1.1; 4.4.6.1; and 4.4.6.2. The plug consists basically of an 80-in. long plug of high density concrete for neutron shielding with space above the shielding for the mechanism.

A single speed reversible, three-phase hollow shaft induction motor is used for positioning the control rod. The motor armature is keyed to the shaft to drive an epicyclic gear train. This gear train gives a speed reduction of 4096:1. The motor speed under full load is approximately 820 rpm. An inner shaft within the hollow shaft is keyed to both the output of the epicyclic gear train and the scram clutch. The purpose of this clutch is to transmit power to the cable drum drive gears. During normal operation the clutch is energized and rod movement, either into or out of the reactor core, is controlled by power to the drive motor. Upon reactor scram, the d-c power to the clutch is interrupted thereby disengaging the clutch and the cable drum, and the drum drive gears are then free to rotate and the control rod, under the force of gravity, will fall into the reactor core. To control scram speeds a retarder is energized by a generator driven through a gear train coupled to the cable drum. During normal rod movements the output of the generator is small and the braking effect is negligible. However, during a scram, as the drum accelerates to high speeds, the generator output increases, in turn increasing the braking force applied to the drum by the retarder. This same gear train drives a camshaft which operates limit switches and position switches which control the application of the braking current to the retarder. No significant drag is exerted on the control rod until it reaches a point 6 ft from the end of its travel. At this point, a position switch is closed to apply generator voltage, through a resistance, to the retarder. The rod motion is retarded to a velocity no greater than 0.7 fps at the fully inserted position, irrespective of the elevation from which it is scrambled.

The crash stop, consisting of a crash arm rotated by a gear train driven by the cable drum, engages a dog on the cable drum to positively stop the control rod at the lower end of its travel. Rod deceleration does not exceed 4 g.

Rod position is indicated by a synchro unit geared into the scram control mechanism ring gear.

A manual drive shaft extends from the top of the control rod drive plug to drive the drum and wind up the cable in the event of malfunction of the motor to permit removal of a faulty drive mechanism. A manual rod lock drive shaft operates a mechanism which locks both the control rod and/or the control rod shroud to the control rod drive plug for removal (Figure 4.4.6.1).

Lead shielding above the drive mechanism shields the area from radiation from the activated control rod drive cable when the cable is wound around the drum.

The control rod drive plug is equipped with a gas tight closure identical to the closure described in Section 5.11.

4.4.7 Insertion and Withdrawal Rates

The control rod drive mechanism inserts and withdraws control rods at the

EGCR Control Rod Drive
Rod-Shroud-Lock Arrangement

FIG. 4.4.6.1

FIG. 4.4.6.2

single speed of approximately two inches per minute.

For a reactor scram the rods descend into the core under the influence of gravity, the net acceleration being approximately 0.6 g. This scram speed has been verified during prototype and environmental testing of the completely assembled control rod drive mechanism.

4.5 Reactor Pressure Vessel

The reactor pressure vessel consists of a pressure-containing shell, an inner temperature barrier, a core support structure, an external support skirt, and a number of nozzles in vessel heads.

4.5.1 Pressure Vessel Design Features

The pressure vessel is shown in Figures 4.5.1.1, 4.5.1.2 and 4.5.1.3. The inside diameter of the pressure shell is 20 ft 0 in. The height inside the heads is 45 ft 8 in. The wall thickness of the cylindrical portion is 2-3/4 in. minimum and of the heads is 4 in. minimum. The vessel is designed for a pressure of 350 psig at 650 F. The operating pressure is 300 psig and the highest operating temperature of the vessel will be below 650 F, probably approximately 575 F when operating at full power. The pressure shell is fabricated of carbon steel, ASTM Designation A-212-57 T Grade B Firebox, with impact properties conforming to ASTM Designation A-300-58, ultrasonically tested.

The vessel and all nozzles are externally insulated. Vessel construction is in accordance with the ASME Boiler and Pressure Vessel Code Section VIII, 1959 edition, together with Addenda of 1960, including Case Interpretations 1270N and 1273N, as well as the Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components, PB 151 987 including Addendum of February 27, 1959. Where the nozzle design was such that radiographic inspection required by the ASME Code Case Interpretations would have been difficult to interpret and would not have full assurance of weld quality, very extensive magnetic particle, liquid penetrant, and ultrasonic inspections were conducted instead.

4.5.2 Vessel Penetrations and Nozzles

The numbers, size, and location of the nozzles which penetrate the top head and the bottom head of the pressure vessel are established by service and operational requirements. The nozzles, extensions and closures are constructed in accordance with the ASME Boiler and Pressure Vessel Code and the Nuclear Code Cases as set forth under Section 4.5.1 above. In addition, all nozzle designs are verified by thermal and stress analysis. All nozzles which terminate outside the reactor vault are double sealed and are equipped with vent connections between the seals to preclude the release of reactor coolant. To maintain continuity of shielding, all nozzles penetrating the biological shield are stepped and fitted with removable shield plugs. Each nozzle which penetrates the biological shield is externally cooled by air flow through the penetration into the reactor vault. Table 4.5.2.1 indicates the quantity, size (at vessel penetration) and service function for the top head nozzles.

All nozzles shown in Table 4.5.2.1, with the exception of the surveillance specimen nozzles, also penetrate the head of the temperature barrier. Each top head nozzle is insulated externally and internally and includes internal liners, and connections for admission of coolant gas to maintain the temperature of the nozzle walls below 650 F. Two burst slug detection system and thermocouple lead nozzles terminate within the reactor vault. Each of these nozzles is penetrated by 117 stainless steel burst slug detection system 1/2-in. OD tubes, 127 stainless steel-sheathed 0.210 in. OD thermocouple leads,

and one 1/2-in. OD PTM pressure tube.

TABLE 4.5.2.1
Top Head Nozzles

<u>Quantity</u>	<u>ID (in.)</u>	<u>Service</u>
25	11-3/8	Control Rods
8	8-1/8	Special Plugs
2	8-1/8	Experimental Loop
1	11-3/8	Experimental Loop
1	11-3/8	Emergency Cooling Loop
2	8-1/8	Surveillance Specimens
2	28	Burst Slug Detection System and Thermocouple Leads
2	34	Reactor Coolant Outlet
2	11-3/8	In-Core Experiments

Table 4.5.2.2 indicates the quantity, size (at vessel penetration) and service function for the bottom head nozzles.

TABLE 4.5.2.2
Bottom Head Nozzles

<u>Quantity</u>	<u>ID (in.)</u>	<u>Service</u>
12	16-7/8	Fuel Charge Nozzles
9	12-3/8	Fuel Charge Nozzles
2	7-3/8	Experimental Loop Nozzles
1	12-3/8	Experimental Loop Nozzles
1	12-3/8	Emergency Cooling Loop
2	11-1/2	Thermocouple Nozzles
2	11-1/2	Thermocouple and Flux Scanner Nozzles
2	22	Reactor Coolant Inlet

Each nozzle penetrating the bottom head is externally insulated. Nozzle cooling is not required. The temperature in this area will not exceed 510 F.

Each thermocouple and flux scanner nozzle terminates within the reactor vault. These nozzles have penetrations as indicated on Figure 4.5.1.1 and in Table 4.5.2.3 below:

TABLE 4.5.2.3
Bottom Head Thermocouple and
Flux Scanner Nozzle Penetrations

<u>Nozzle Location</u>	<u>Number of 1/8-in. Thermocouple Leads</u>	<u>Number of 3/8-in. Flux Scanner Leads</u>
SE	24	9
NE	36	0
NW	31	0
SW	31	9

Two PTM probes from the top plenum leave through each of the two top head BSD nozzles. The two pressure probes from the bottom plenum leave through bottom head thermocouple nozzle No. 3 (NE). A manhole in the reactor coolant piping affords temporary access to the vessel plenums prior to startup.

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undesirable conditions: the pressure drop across the top fuel assembly would approach the value at which the assembly would float as the column is lowered out of the core, and the graphite column would experience unnecessarily high temperature gradients over its cross-section due to the increased coolant flow. Some over-cooling of the graphite is unavoidable when the annulus seal is broken and annulus flow increases considerably, but in general it is desirable to minimize over-cooling. Annulus exit coolant temperature is 750 F for this condition compared with 1050 F for a normal fuel channel.

A minimum flow is established in an empty fuel channel to give approximately 750 F outlet coolant temperature. Thus, graphite column temperature gradients would be about the same as those encountered during the initial part of the unloading cycle. The required flows during loading operations, as a function of position of the fuel column, are the same as during unloading.

Unloading of fuel assemblies with the service machine is permissible only at or below 20% reactor power and flow. The pressure drop across the reactor core corresponding to 20% power and flow is not large enough to cause lifting of a fuel assembly.

A fuel assembly removed from the core is held in the fuel channel extension at the top of the core for ten minutes to allow fission products to decay before transferring the assembly into the service machine. After this holding time, the after-heat has decayed to a sufficiently low level that the fuel element cladding temperatures remain below 1600 F, with all heat being transferred by natural convection and radiation to the surroundings.

4.7 Nuclear Design Features

4.7.1 Neutron Source

The neutron source for the EGCR is located in control rod position P (see figure 4.3.2.1). The neutron source assembly, very similar in shape and size to a control rod, consists of a control rod adapter, a long section of 2-1/2 schedule 10 stainless steel pipe, and a source support tube in which is encased a beryllium sleeve and an antimony-oxide pin. (The EGCR neutron source assembly is shown in figure 4.7.1) The neutron source assembly is attached to a regular EGCR control rod drive, and is used with a control rod shroud.

The neutron source assembly is slightly shorter than a regular control rod so that when the neutron source assembly is in the fully withdrawn position, the centerline of the beryllium sleeve and the antimony-oxide pin is 1 foot 9 inches above the top of the top graphite reflector. When the neutron source is fully inserted, the centerline of the beryllium sleeve and antimony-oxide pin is 3 feet 1 inch below the centerline of the core. The normal operating position for the neutron source will be in the fully inserted position; it will be withdrawn only when reactor period measurements are being made. The withdrawal ensures that the peak power during the measurement remains below the experimentally accepted level.

The neutron source assembly is cooled in the same manner as the control rods. The antimony charge is antimony-tetroxide (SB_2O_4) which is stable

EGCR NEUTRON SOURCE ASSEMBLY
FIG. 4.7.1

to 1,700 F. The beryllium sleeve is made of hot-pressed QVM beryllium. The remainder of the neutron source assembly is made of stainless steel.

When the reactor is shut down by the largest margin ($k = 0.85$), the neutron source gives an adequate count rate in the start-up chambers which are located in the thermal columns.

4.7.2 Control Rod Worth

The accuracy of EGCR control rod worth calculations was determined by comparing the calculated results with experimental values obtained at Hanford using the Physical Constant Test Reactor (PCTR).⁽³²⁾ In this experiment, a 16-cell array of EGCR lattice cells was assembled and the multiplication factor of the assembly with a reflective boundary condition was measured with a control rod inserted at the center. In this condition the assembly was equivalent to an infinite array of control rod supercells with 16 lattice cells per supercell.

The results of this experiment verified the accuracy of the previously adopted four-group model of the control rod. In this model the control rod is assumed to be black to all neutrons below 4.5 ev, constituting the lowest two groups of the calculation, and diffusion theory is applied in the upper two groups. For geometrical simplicity the cylindrical rod was replaced in parts of the analysis by a rod of square cross section with the same absorption area according to the definition of Hurwitz and Roe.⁽²⁴⁾ The results of the comparison calculations⁽³³⁾ for heterogeneous and homogeneous models for the lattice cells and for both square and round cross sections are given in Table 4.7.2. A comparison of the calculated and measured thermal flux distributions is shown in Figure 4.7.2.1 for two traverses through the supercell.

TABLE 4.7.2
Comparison of Calculated and Experimental Values
of the Worth of EGCR Control Rods in a
16-Lattice Supercell

<u>Description</u>		<u>Supercell</u> <u>Geometry</u>	<u>Control Rod</u> <u>Δk</u>
<u>Control Rod</u> <u>Geometry</u>	<u>Reactor Lattice</u> <u>Region</u>		
Square	Heterogeneous	Square	0.152
Square	Homogeneous	Square	0.154
Cylindrical	Homogeneous	Cylindrical	0.150

PCTR experimental value: $k = 0.158 \pm 0.012$

The calculations confirmed the accuracy of the control rod model, and the control rod worth calculations for the reactor followed the second line of Table 4.7.2. The core was treated as a homogeneous medium, and the control rods were individually represented by square regions. Two-dimensional calculations of the entire core were performed in this manner so that the effect of leakage, for example, on rod worth was explicitly included. The shutdown worth of the full 21-rod bank in the cold reactor was calculated to be $\Delta k = 0.241$. Figure 4.7.2.2 shows the variation of Δk with insertion of a full 21-rod bank. Figure 4.7.2.3 shows the Δk per inch for various depths of insertion of a full 21-rod bank.

4.7.3 Reactivity Requirements

The fuel enrichment for the initial core loading was determined by fuel cycle economics by the requirement for adequate shutdown margin at highest reactivity.

FIG. 4.7.2.1 COMPARISON OF MEASURED AND CALCULATED THERMAL FLUX DISTRIBUTIONS IN PCTR CONTROL ROD EXPERIMENT

FIG. 4.7.2.2 Δk vs DEPTH OF INSERTION FOR FULL 21-ROD BANK OF CONTROL RODS

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FIG. 4.7.2.3 Δk PER INCH OF WITHDRAWAL vs DEPTH OF OF INSERTION FOR FULL 21-ROD BANK OF CONTROL RODS

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installed and to seal the reactor gas during the interim period prior to completion of the loop. This closure will support shield plug or an instrumented fuel assembly. When the loops are installed, the closure will support the shield plug and the loop test specimen.

To permit access to the welded breech plug in the bottom nozzle inner tee section, the containment has a ball lock and crush-type O-ring seal backed up by a double O-ring seal and flange located at the charge machine ceiling. A bottom shield plug and ball-lock assembly are handled using the charge machine and a special adapter.

d. Interior Nozzle Cooling System

The purpose of the nozzle cooling system is to control the temperature of the top and bottom nozzle outer tees and the reactor vessel nozzles and nozzle extensions. The through-tube nozzle cooling system is part of the vessel cooling system provided for the reactor to cool the gas to the charge machine, service machine, reactor nozzles, and the reactor vessel. Helium is extracted from the inlet duct of the reactor coolant system which is normally operating at 508 F. The fluid passes through one of the two compressors of the system where the pressure is raised to overcome system resistance. The fluid then flows through a cooler where the temperature is reduced to 125 F. The helium then flows to a header distribution system where it is distributed to each component. The coolant for the top nozzle enters below the inner tee and flows down the nozzle between the tube and the reactor pressure vessel nozzle extension, discharging into the upper reactor plenum.

e. Leak Detection System to Top and Bottom Nozzle Outer Tees

Three seals are located at the top nozzles. The first seal is designed as part of the breech closure and seals reactor gas from the secondary containment when the through-tube is not installed. With the through-tube installed, it seals loop gas from the containment piping. Double O-rings on the blind flange seal the containment piping. Two connections for leak checking the seals are provided. One connection is located between the breech seal and the first O-ring on the blind flange and is designated as the closure area, while the other connection is located between the two O-rings on the blind flange and is designated as the double O-ring area. The double O-ring area line has a valve that is normally closed and a pressure indicator upstream of the valve. This line is connected to the leak-off and recovery system. The closure area line, once away from the nozzle, has a tee connection that contains a valve and is connected to a vacuum system. This valve is normally closed. The line downstream of the tee has a valve that is normally opened with a pressure indicator connected in the line. This line also connects to the leak-off and recovery system.

In principle, the bottom seals are identical to those at the top except that the closure seal has reactor gas instead of loop gas on the inside at all times regardless of whether or not a through-tube is installed. Both top and bottom blind flange seals must be connected to the leak-off and recovery system before reactor startup in order to monitor the seal areas.

4.8.5 Surveillance Specimens

Surveillance specimens are installed in the reactor vessel to monitor the effects of radiation on the reactor materials. The surveillance specimens are reactor pressure vessel steel specimens and graphite specimens.

a. Pressure Vessel Irradiation Specimens

The EGCR pressure vessel is constructed of carbon steel, type SA-212, grade B. The highest dose for the reactor vessel is in the region near the horizontal midplane where the fast-neutron flux will accumulate to approximately 10^{18} nvt. The temperature in the region is approximately 550 F. There are at present no test data on such material which are directly applicable to the EGCR design. Of particular interest is the embrittling effect of irradiation. A number of surveillance specimens are to be installed in tubes attached to the surveillance nozzles. These tubes terminate in the annulus between the thermal barrier and the pressure vessel, where the values of the flux and temperature are near those which the vessel will experience. Removal of the specimens is effected through the surveillance nozzles. The number of specimens and the proposed conditions of the test are tabulated in Table 4.8.5. In addition to the samples that are irradiated in the reactor, a number of samples are to be provided from the vessel material for out-of-reactor testing so that information from in-pile material tests can be correlated with data obtained from the out-of-pile tests.

b. Graphite Specimens

To monitor irradiation effects such as irradiation-induced shrinkage, buildup of stored energy, and oxidation rates, a number of graphite samples will be placed in the core. These samples will be removed and examined during the lifetime of the plant. The specimens will be placed in central channels, roughly in the same position as the graphite fuel assembly sleeves. The temperature at the specimens will be 500, 750, and 1150 F at the bottom, middle, and top positions respectively.

4.8.6 Flux Scanners

It is expected that the power distribution in the EGCR may change markedly from time to time, both as a result of changes in the control rod pattern and, in the future, as a consequence of experimental loop operation. The need to know the power distribution arises from the need to control the coolant flow distribution. This control is effected by means of the adjustable orifice in each fuel channel. Orifices will be adjusted to yield the same gas outlet temperature from all fuel channels. The flux scanner system is not a part of the reactor safety system, nor the essential process instrumentation. It is an operational aid which will greatly simplify the task of adjusting flow orifices and will facilitate the interpretation of various experiments that may be performed in the reactor. Since adjustment of orifices over much of the reactor core will be a very laborious and time-consuming operation, and since orifice adjustments will not be independent (that is, flow in one channel will be affected by orifice adjustments in a large number of other channels), it is necessary to predict orifice settings before undertaking large-scale readjustments. The flux-scanner system will provide flux plots of the reactor core which can be used to determine core heat generation and required coolant flows.

→ Thin-walled stainless steel tubes penetrate the core axially in 18 positions, each equidistant from the four adjacent fuel channels. Wires containing manganese are driven into the core, activated, withdrawn, and scanned sequentially by one of three scintillation counters. The quantity of principal interest will be the integral of the flux distribution over the length of the wire spanning the full height of the core. Any variations greater than a few percent that may occur in axial distributions will be detectable. From the 18 flux readings it will be possible to confirm or correct calculated flux distributions.

TABLE 4.8.5

Pressure Vessel Surveillance Specimens

<u>Specimen Taken From</u>	<u>Exposure Time in Reactor (year)</u>	<u>Radiation Exposure (nvt)</u>	<u>Number of Charpy V Specimens</u>	<u>Number of Subsize Tensile Specimens</u>
Base Metal	2	10^{17}	20	3
Base Metal	5	2.5×10^{17}	20	3
Base Metal	10	5.0×10^{17}	20	3
Base Metal	15	7.5×10^{17}	20	3
Base Metal	18	9.0×10^{17}	20	3
Base Metal	20	10^{18} (c)	20	3
		(c)	12	2
Weld Metal	2	10^{17}	10	3
Weld Metal	5	2.5×10^{17}	10	3
Weld Metal	10	5.0×10^{17}	10	3
Weld Metal	15	7.5×10^{17}	10	3
Weld Metal	18	9.0×10^{17}	10	3
Weld Metal	20	10^{18} (c)	10	3
		(c)	20	2
RPI HAZ (b)	(a)		36	
Total			248	40

(a) When base metal data indicate damage, specimens are withdrawn.

(b) Specimens are taken from the base metal and given the RPI treatment to simulate material in the heat-affected zone.

(c) These specimens are to be exposed for a 20-year period or may be withdrawn at an earlier time to be used in annealing tests if the data obtained from earlier tests indicate the need for annealing the pressure vessel to restore the impact strength of the pressure vessel.

Each of the 18 tubes penetrating the reactor pressure vessel is routed through the lower grid structure of the reactor to the appropriate position beneath the graphite stack, and then passes up through the core. The tubes are closed at the upper ends; the insides of the tubes are at atmospheric pressure. The tubes
→ are fabricated from type 304 stainless steel and are 0.375 in.OD and 0.260 in.
→ ID. They are designed to withstand 350 psi external helium pressure at reactor
→ core operating temperatures.

→ In addition, six ion chambers may be inserted into the core via the high-power
flux wires and the channels traversed at slow speed. Flux levels are traced on
→ chart recorders located in the main control room.

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minimize the radiation streaming through these gaps, collars are attached to the nozzles and the nozzle diameters are increased in discrete steps.

Each of these top and bottom biological shields consists of a steel grid structure filled with high density concrete. The concrete serves only as a shield since the steel grid forms the structural member.

Ten feet of conventional concrete is used in the vertical biological shield walls around the reactor vessel. The inside surface of the biological shield is continuously cooled by the shield cooling air. A vertical gap one inch wide, filled with resilient joint material, accommodates thermal expansion of the inner four feet of the shield wall which absorbs most of the heating. All ducting through the biological shield makes two 90 deg bends to reduce radiation streaming from the reactor vault.

5.2.3 Shield Plugs

Shield plugs are inserted into all the reactor vessel access nozzles, both above and below the reactor to prevent streaming of neutrons and gamma rays through the biological shield, and thereby permit personnel access to the service machine and charge machine areas. High density ferrophosphorous concrete is used in all shield plugs.

5.2.4 Coolant System

The reactor coolant system shielding consists of ordinary concrete walls separating the outlet and inlet helium pipes, the reactor coolant blowers, and the steam generators from adjacent areas. The thickness of these walls is based on the assumption that 1000 curies of fission products with an effective gamma energy of 1 Mev per disintegration are dispersed in the reactor coolant system. The shielding for the reactor coolant piping is designed on the assumption that the 1000 curies are uniformly distributed throughout the system. The steam generator shielding thicknesses are designed on the assumption that the 1000 curies are concentrated within the steam generator.

5.2.5 Control Room

Concrete shielding is installed between the control room and the reactor building to permit control room occupancy following an accident. The main control room area containing the main control panels is shielded by a 2 ft 6 in. thick wall between the room and the reactor building. The adjacent auxiliary equipment room has an 8 in. thick concrete block wall between the room and the reactor building. The building structure serves as remaining required shielding.

5.3 Reactor Containment

5.3.1 Containment Philosophy, Criteria and General Description

The containment system for the EGCR serves to protect the public against radiation exposure in excess of established safe values by limiting the activity which can be released by any credible occurrence. In general, dispersal by atmospheric discharge is permissible up to the limit of public exposure (Section 9.1).

The building is designed not only to contain reactor coolant which would be released following a coolant system failure, but also to contain nuclear and chemical energies which could be released. The maximum release does not result in internal pressures greater than 9 psig or in shell temperatures in excess of 200 F.

The containment system includes a steel pressure vessel which surrounds the reactor complex.

Numerous penetrations are necessary for electrical and instrumentation leads for heating and ventilation ducts, and for the connection of the reactor complex to its auxiliary systems and to the turbine-generator system. Isolation valves installed in certain lines penetrating the shell complete the isolation system. These valves require closure in the event of abnormal or accidental release of radioisotopes from the enclosed reactor system.

A criterion for plant design requires that the reactor operator should not be relied upon to take action from the control room within thirty minutes in the event of an accident or forty-five minutes if corrective action requires the operator to leave the control room. Therefore, the activity release during these periods must not result in excessive doses outside the containment building. Any release, either controlled, abnormal, or accidental, which would otherwise exceed the established safe level must be prevented by automatic operation of the containment isolation system (Section 9).

The containment building surrounding the reactor complex is a 114-ft diameter cylindrical vessel with a hemispherical top head and a torospherical bottom head. The inside height is approximately 216 ft. The bottom head, approximately 70 ft below grade, is encased in concrete. The cylindrical portion contains nozzles and penetrations required for ventilation, interconnection of systems, and two air locks, one for equipment or personnel and one for personnel only.

The containment building is designed to the ASME code ⁽¹⁾ for an internal pressure of 9 psig at 200 F. The shell and head plates are constructed of SA-201, Grade B steel to A-300 requirements. All pressure stressed welds, welds attaching penetrations to the shell, and welds which, if defective, would represent a weakness in the structure are 100% radiographed except that magnetic particle or fluid penetrant inspections are used in the case of welds not amenable to radiography. The entire structure is pressure tested to 11.25 psig with the air locks in place, and is leak tested at 9 psig after completion of all penetrations. A 96-hr leakage test was conducted on the containment shell during April 1962. The leakage rate was determined to be less than 0.1% per 24 hr.

5.3.2 Containment Building Design

The design of the containment building is in conformance with the following ASME Boiler and Pressure Vessel Code sections:⁽¹⁾

- a. Section II, Material Specifications
- b. Section VIII, Unfired Pressure Vessels
- c. Section IX, Welding Qualifications.

In addition, the following ASME Nuclear Code Cases apply:

- a. 1270N, General Requirements for Nuclear Vessels
- b. 1271N, Safety Devices
- c. 1272N, Containment and Intermediate Containment Vessels

The following tabulation lists the loads, pressure, and temperature used in the design:

Snow load	10 psf
Insulation	1 psf
Basic wind pressure, top head	40 psf
Basic wind pressure, cylinder	30 psf

The leakage rate tests are conducted using the absolute temperature-pressure method employing a water column approximately 30 ft high and located outside the containment shell to measure shell pressure. A system of thermocouples is installed inside the containment shell to measure the average air temperature and these readings are used to correct the shell pressure for variations in air temperature. The air within the containment shell is stratified and thermocouples are located so that each represents a known volume of air and fans provide local circulation at each thermocouple location.

The relative humidity of the shell air is measured during the leak test to detect possible excess and resultant condensation in any part of the shell. Any condensation during the test renders the results invalid. The pressure is determined by reading the water column to the nearest 0.1 in. H₂O, correcting for temperature, and adding the barometric pressure converted to inches of water.

Retest procedures require the reactor and auxiliary systems to be shut down. Equipment or instrumentation presently installed in the plant will not be damaged by retest pressurization. Pressure retest procedures are described in Volume II under Section 17: Periodic Plant Tests and Inspections.

5.3.6 Missile Protection

Missile generation was carefully considered during the design of the EGCR. During a release of reactor coolant, containing activity, it is imperative that the containment building remain intact. No missiles generated as the result (or cause) of the release may impinge on the containment building, directly, or by ricochet from walls or equipment in the forward direction of flight.

The containment building could only be damaged by missiles in certain regions above the ground floor. The major items of the high pressure helium systems are either below grade or enclosed within concrete rooms. The principal exception is the service machine area, which is enclosed by concrete walls on all sides but is open above so that the top head of the containment building is exposed to potential missiles from the reactor vessel top nozzles, the service machine, and associated piping systems, and from the blower seal water system.

Potential missiles which could be generated during a release of reactor coolant include valve stems, thermocouple wells, pressure instrumentation, blind flanges, hatch covers, doors, shield plugs, fuel assemblies, instrumented fuel columns, control rods, special unlocking tools, and the fuel assembly grapple.

The metal thicknesses of the service machine vessel and its associated shielding assure adequate protection against any missile which can be generated in or under the machine.

All potential missiles associated with the exterior of the service machine and with its piping systems, such as valve stems, instruments and thermowells, are directed away from the exposed building shell.

A number of preventive features of the EGCR design greatly reduce the chances of generating a missile:

- a. All of the helium service lines connecting directly to the interior of the service machine, have check valves installed near the service machine pressure vessel. This limits the rate of depressurization of the service machine in the event of a line rupture. In the helium supply lines from the vessel cooling system two excess flow valves

are installed in the upstream side of each flexible section. These parallel supply lines preclude the possibility of one valve accidentally shutting off the flow of cool helium to the service machine.

- b. When the service machine touches the top nozzle flange, two limit switches on the service machine extension nozzle de-energize the service machine bridge and trolley and operate the brakes on the crane. The service machine seismic restraints also are engaged at this time. These features prevent inadvertent movement of the service machine while it is connected to a reactor vessel nozzle.
- c. The service machine closure valve is interlocked to prevent opening the closure valve when a pressure differential exists across the valve. If the interlock should fail, the valve still cannot open because the drive motor is not of sufficient capacity to meet the torque requirement with pressure differential across the valve. The motor would trip out on overload.
- d. Interlocks also prevent unlocking a top nozzle closure if a pressure differential exists across the top nozzle plug.
- e. The service machine is designed, fabricated and tested in accordance with the ASME Boiler and Pressure Vessel Code (see Section 5.10). The materials of construction for the service machine have nil-ductility transition temperatures below the minimum expected ambient conditions, including adequate allowance for the effects of integrated radiation damage over the expected 20-year service life. This effectively precludes the possibility of brittle rupture and subsequent fragmentation of the service machine pressure vessel. Thus, no secondary missiles in the form of vessel fragments need be considered.
- f. The service machine pressure vessel is protected against overpressurization by a relief valve. Further, during the period when the nozzle plug is removed from a top nozzle the machine constitutes an effective portion of the reactor coolant system. During this time, it also is protected against overpressurization by the three relief valves on the main reactor coolant system.
- g. The removable sections of the service machine shielding (lead canned in steel) are securely attached to the structure surrounding the service machine so that they cannot become missiles in the event of a sudden release of helium.
- h. Multiple sets of contacts for each nozzle closure assure positive indication of closure seating and rotation even if one set of contacts should stick or be damaged.
- i. The service machine drive units are enclosed in housings external to the service machine pressure vessel. These housings constitute an integral part of the service machine pressure vessel and are designed to the same code requirements. Therefore, rapid depressurization of the service machine cannot result from a seal or gasket failure on the drive shaft.

Other potential missiles in the region of the top head of the containment building are the valve stems on the high pressure helium lines to the blower seal water head tanks. These valves are oriented so that the valve stems are not pointed toward the shell or the head tanks.

The only remaining region where the building is exposed to potential missiles from high pressure helium piping is at the vessel cooling system headers and

flow panels. The valves in this region are also safely oriented.

Hatches and doorways are located so that the containment building shell would not be hit directly by pieces of equipment generated as missiles within any enclosed area. Hatches and doors are located so that they could not hit the shell directly if they were to be blown off due to a sudden pressure surge within rooms containing high pressure helium equipment or piping.

The anchorage of equipment both inside and outside the containment shell has been investigated to assure that this equipment is properly anchored and cannot generate a missile which would penetrate the containment shell.

5.3.7 Access

Access to the containment building is through two air locks shown on Figure 5.3.7. The equipment air lock is 11 ft in diameter, capable of accommodating equipment items up to 50 ft in length; the other air lock is 7 ft in diameter for personnel use.

The air locks are fitted with pressure doors which swing away from the barrel, interlocks, pressure equalizing valves, gaskets, concrete floor, removable decks, interior lighting, and all necessary equipment to make it operable under design pressure and temperature.

The doors of each air lock are interlocked to allow operation of only one door at a time from either side of the door, when pressure is equalized on the two sides of the door. A mechanical linkage with the door locking mechanism automatically operates the pressure equalizing valves. The door interlocks can be over-ridden under controlled conditions. The override is key operated to prevent inadvertent or unauthorized operation of the over-ride feature. The door opening device is a hand pull.

Indicator lights are installed at two locations for each air lock door; one set is visible from within the containment building, and the second set is visible from outside of the exterior door. Each set of indicator lights indicates "door closed and locked" and "door unlocked" for each door. Duplicate indicator lights are installed in the main control room.

Each air lock contains a telephone, alarm horn switch and an emergency battery powered light which is turned on by a failure of the normal lighting system.

In the event of a serious accident which might pressurize the interior of the containment shell, personnel inside can get out of the shell via either the personnel or equipment air lock. The time necessary to go through the personnel air lock with the shell under 9 psig pressure is approximately four minutes, and through the equipment air lock approximately six minutes. These times include the time required to turn the hand wheels, equalize pressure, open and close both air lock doors.

A spiral type of emergency personnel escape chute is located adjacent to the personnel air lock to facilitate evacuation of personnel from within the containment building.

5.3.8 Penetrations and Isolation Provisions

The containment shell penetrations and their isolation provisions are listed in Table 5.3.8. These penetrations can be grouped into the following classifications:

a. Automatic Containment Isolation

These penetrations include lines which vent directly to the atmosphere surrounding the containment shell, and lines which provide nonessential services. The fourteen penetrations which vent directly to the atmosphere are each provided with two isolation valves in series.

b. Remote Manual Isolation

These penetrations are those providing essential plant services such as main feedwater lines, main steam lines, instrument air, service water, helium supply from the outside helium storage tanks, and carbon dioxide from outside storage.

c. Normally Closed or Check Valves

Penetrations in this classification include electrical and instrumentation, the fuel discharge chute, equipment air lock, personnel air lock, the bottom blowdown lines and miscellaneous drains from the steam generator safety valves, decontamination lines, one containment shell vacuum relief valve and the capped sump drain.

d. Terminate in a Closed System

Seven of the eight experimental loop openings and six of the seven rupture discs between the experimental cells and the containment shell are blanked off. ←

TABLE 5.3.8

<u>Penetrations in Containment Shell</u>			
<u>System or Service</u>	<u>Number of Penetrations</u>	<u>Penetration Closure Classification</u>	<u>Valve Size & Type</u>
Ventilation Intake Charge Machine Room	1	<u>Automatic</u> , fails closed on loss of electric power or instrument air, two valves in series	ECV-37-2, ECV-37-4 36-in. butterfly
Ventilation Intake Blower Motor Room	1	"	ECV-29-1, ECV-29-7 30-in. butterfly
Ventilation Intake Blower Motor Room	1	"	ECV-29-4 ECV-29-10 30-in. butterfly
Ventilation Intake Biological Shield Cooling Air	1	"	ECV-27-2, ECV-27-6 24-in. butterfly
Ventilation Intake Reactor Building and Service Machine Room	1	"	ECV-41-1, ECV-41-4 42-in. butterfly
Ventilation Discharge from Charge Machine Room	1	"	ECV-37-1, ECV-37-3 36-in. butterfly

TABLE 5.3.8 (continued)

Penetrations in Containment Shell

<u>System or Service</u>	<u>Number of Penetrations</u>	<u>Penetration Closure Classification</u>	<u>Valve Size & Type</u>
Ventilation Discharge from Steam Generator Room	1	<u>Automatic</u> , fails closed on loss of electric power or instrument air, two valves in series	ECV-29-2, ECV-29-8 20-in. butterfly
Ventilation Discharge from Steam Generator Room	1	"	ECV-29-3, ECV-29-9 20-in. butterfly
Ventilation Discharge Biological Shield Cooling Air	1	"	ECV-27-3, ECV-27-7 42-in. butterfly
Ventilation Discharge Service Machine Room	1	"	ECV-41-2, ECV-41-5 36-in. butterfly
Sanitary Sewer	1	"	ECV-31-1, ECV-31-2 4-in. piston operated ball valve
Hot Waste	1	"	ECV-19-1, ECV-19-20 4-in. diaphragm operated ball valves
Warm Waste	1	"	ECV-19-10, ECV-19-21 4-in. diaphragm operated ball valves
Gas vent to stack and diversion	1	<u>Automatic</u> , diverts to reactor building on loss of electric power or instrument air, two-3-way valves in series	ECV-14-32, ECV-14-42 10-in. 3 way valves
Continuous Blowdown from Steam Generator C-1	1	<u>Automatic, nonessential service</u> , fails closed on loss of electric power or instrument air	ECV-10-13, 1-in. diaphragm operated valve
Saturated Steam Sample from Steam Generator C-1	1	"	ECV-10-16 1-in. diaphragm operated valve
Continuous Blowdown from Steam Generator C-2	1	"	ECV-10-12, 1-in. diaphragm operated valve

TABLE 5.3.8 (continued)

<u>Penetrations in Containment Shell</u>			
<u>System or Service</u>	<u>Number of Penetrations</u>	<u>Penetration Closure Classification</u>	<u>Valve Size & Type</u>
Saturated Steam Sample from Steam Generator C-2	1	<u>Automatic, nonessential service, fails closed on loss of electric power or instrument air</u>	ECV-10-17 1-in. diaphragm operated valve
Helium Discharge to Storage Tanks	1	"	ECV-17-9, 1-in. diaphragm operated valve
Domestic Water	1	"	ECV-21-12, 4-in. diaphragm operated valve
Fire Protection Water	1	"	ECV-21-13, 6-in. diaphragm operated butterfly valve
pH Control Chemical Feed to Steam Generator C-1	1	"	ECV-8-26 1-in. diaphragm operated valve
pH Control Chemical Feed to Steam Generator C-2	1	"	ECV-8-27, 1-in. diaphragm operated valve
Make-up to Experimenters' Cooling Water	1	<u>Remote manual, fails "as-is"</u>	ECV-26-47, 1-in. diaphragm operated valve
Building Heating Hot Water Return	1	<u>Automatic, nonessential service, fails closed on loss of electric power or instrument air</u>	ECV-24-35, 4-in. diaphragm operated valve
Building Heating Hot Water Supply	1	"	ECV-24-36, 4-in. diaphragm operated valve
Plant Air	1	"	ECV-32-9, 3-in. diaphragm operated valve
Helium Supply from Storage Tanks	1	<u>Remote manual, essential service, fails open</u>	HCV-15-1, 1-in. diaphragm operated valve
Helium Supply from Storage Tanks to Service Machine	1	<u>Remote manual, essential service, fails open, closes automatically on low reactor coolant pressure</u>	HCV-22-11, 1½-in. diaphragm operated valve

TABLE 5.3.8 (continued)

Penetrations in Containment Shell

<u>System or Service</u>	<u>Number of Penetrations</u>	<u>Penetration Closure Classification</u>	<u>Valve Size & Type</u>
→ Service Water	1	<u>Remote manual, essential service</u> , fails open	HCV-5-36, 12-in. double acting piston operated butterfly valve
→ Service Water	1	<u>Remote manual, essential service</u> , fails open	HCV-5-37, 12-in. double acting piston operated butterfly valve
Service Water	1	"	HCV-5-40, 12-in. double acting piston operated butterfly valve
Service Water	1	"	HCV-5-39, 12-in. double acting piston operated butterfly valve
Instrument Air	1	"	HCV-6-1, 2-in. diaphragm operated valve
Instrument Air	1	"	HCV-6-2, 2-in. diaphragm operated valve
→ Main Steam Pipe from Steam Generator C-1	1	<u>Remote Manual, essential service</u> , fails "as-is", automatically closes upon steam generator isolation	HSV-3-48, HSV-3-182 6-in. motor operated valves
Main Steam Pipe from Steam Generator C-2	1	"	HSV-3-49, HSV-3-181 6-in. motor operated valves
Feedwater Line to Steam Generator C-1	1	"	HSV-3-33, HSV-3-179 4-in. motor operated valves
Feedwater Line to Steam Generator C-2	1	"	HSV-3-32, HSV-3-180 4-in. motor operated valves
→ Carbon Dioxide from storage	1	<u>Remote Manual, essential service</u> fails open	HCV-30-1, 2-in. diaphragm operated valve
Electrical and Instrumentation leads	24, 24-in. large openings	<u>Normally closed</u>	No valves

TABLE 5.3.8 (continued)

<u>Penetrations in Containment Shell</u>			
<u>System or Service</u>	<u>Number of Penetrations</u>	<u>Penetration Closure Classification</u>	<u>Valve Size & Type</u>
Spent Fuel Discharge Chute	1, 18-in. tube	<u>Normally closed, interlocked valves</u>	18-in. motor operated gate valve at upper end 18-in. quick acting pneumatic lock at lower end
Personnel Air Lock	1, 7-ft diam.	<u>Normally closed, interlocked doors</u>	See Section 5.3.7
Equipment Air Lock	1, 11-ft diam.	<u>Normally closed, interlocked doors</u>	See Section 5.3.7
Sump Drain	1	<u>Normally closed, capped drain</u>	No valves
Vacuum Relief for Containment Shell	1	<u>Normally closed or check valves</u>	One vacuum relief valve
Vent from Steam Generator C-1 & C-2 Safety Valves	2	<u>Normally closed</u>	PSV-3-70, PSV-3-39, PSV-3-80, PSV-3-51. Safety Valves
Drain from safety valves of Steam Generators C-1 and C-2	1	<u>Normally closed, by steam generator safety valves</u>	-
Bottom Blowdown from Steam Generator C-1	1	<u>Normally closed</u>	1-in. V - 1231 ½-in. V - 1231, manually operated stop valves
Bottom Blowdown from Steam Generator C-2	1	<u>Normally closed</u>	1-in. V - 1231, ½-in. V - 1231, manually operated stop valves
Miscellaneous drains from Steam Generator C -1	1	<u>Normally Closed</u>	3-in V-1247, and 3-in. V-1244, 3-in. check and stop valves in series
Miscellaneous drains from Steam Generator C-2	1	<u>Normally closed</u>	"
Decontamination Line	1	<u>Normally closed</u>	4-in. manually operated valve
Decontamination Line	1	<u>Normally closed</u>	2-in. manually operated valve

TABLE 5.3.8 (continued)

Penetrations in Containment Shell

<u>System or Service</u>	<u>Number of Penetrations</u>	<u>Penetration Closure Classification</u>	<u>Valve Size & Type</u>
Experimenters' Cooling Water	1	<u>Normally closed or check valve</u>	6-in. V-204, 6-in. check valve
→ Experimental Loops	8	<u>Terminate in a closed system</u>	7 42-in. bellows jointed blank flanged duct → 1 42-in. bellows jointed duct
→ Experimenters' Cell	8	<u>Terminate in a closed system</u>	6 rupture disc pressure relief device blanked off. 1 rupture disc pressure relief device, Cell No. 1. 1 blank flange, Cell No. 8.
→ Fission Product Removal System	1	<u>Normally closed Remote Manual</u>	2 4-in. vent valves in parallel to stack.

Closure of those penetrations which are classified under "automatic containment isolation" is initiated by high gaseous activity in the stack effluent, low reactor coolant system pressure, and manual actuation of switch S-1. The control circuitry for these automatic closure valves is shown in Figures 5.3.8.1 and 5.3.8.2. The relay contacts on these drawings are drawn in the de-energized condition. For those penetrations which have two valves in series, the circuitry to each valve of the pair is independent from the point of coincidence on.

Automatic closure on low pressure in the reactor coolant systems is effected on coincidence of two out of three pressure detectors. The set point for low pressure isolation is 225 psig in the intermediate and power ranges. When the main reactor control mode selector switch is in the start-up position the low pressure isolation is bypassed. When two out of three coincidence is obtained from the pressure detectors the relay contacts in the low reactor pressure safety system mixer stages PQ-12-409A and PQ-12-410A will open. One of these mixer stages receives its power from failure-free motor generator No. 1 and the other from motor generator No. 2. The trouble contacts for the low reactor pressure system mixer stages PQ-12-409A and PQ-12-410A are arranged in parallel. Thus, either motor generator No. 1 or No. 2 power source can fail and not cause a false containment isolation.

The two stack monitors each have detectors for total gaseous gamma activity. These detectors are utilized in a two out of three coincidence matrix to isolate containment on high stack activity. The coincidence requirement is such that containment isolation will be initiated on high stack activity from two of the detectors located in the same monitor. Loss of two of three separate power sources supplying the three gaseous activity monitors used for isolation would have to occur before false containment isolation would be effected.

The closure time of the automatic containment isolation valves is two seconds or less. These valves fail closed on loss of either instrument air or electrical power to the solenoid valves.

The individual valves are equipped with lights which indicate in the control room whether the valve actuator is in the "open" or "closed" position. In addition, an alarm is sounded in the main control room if any of the automatic isolation valves are "not fully open" or "not fully closed" (following an isolation signal). The evacuation alarm system gives an audible alarm in the reactor building, service building, turbine building, and control building whenever containment isolation valves are closed.

The operator receives various information on which to evaluate whether containment isolation should be initiated by using switch S-1. High stack activity alarms are received for particulate, gaseous, and iodine activity in the stack effluent. There is a common alarm for high activity from building particulate material. Similarly, there is a common alarm for high area activity in the building due to gamma radiation. Two activity monitors are located in the cold legs of the reactor coolant system and give an alarm on high coolant activity. The operator has available in the control room numerous other items of information and alarms which enable him to assess whether the seriousness of a particular situation warrants manual closure of the containment isolation valves.

Switch S-1 is a two-position switch, which in the open position, may be overridden by the automatic closure signal, or by the hand controls when desired.

For the double-valved closures, a common hand switch operates both valves during manual control. The hand control switches for these double-valved closures are five-position switches marked as follows:

- a. "close 1 & 2"
- b. "close 1"
- c. "automatic"
- d. "close 2"
- e. "close 1 & 2"

For testing or manual operation, the individual valves may be operated by selection of the desired switch functional position. The automatic closure control function will operate regardless of the individual hand switch positions. After an automatic closure action, all hand switches must be in the "automatic" position in order to open the valves. All valves will then open together when the reset button is actuated (see coordinates C-2 on Figure 5.3.8.1) if the automatic isolation signal source has been removed.

The hand switch for the vent diversion valves, ECV-14-32 and ECV-14-42, is a five-position switch with the following functions:

- a. "divert 1 & 2 to shell"
- b. "divert 1 to shell"
- c. "automatic"
- d. "divert 2 to shell"
- e. "divert 1 & 2 to shell".

→ The switch controlling the single automatic containment isolation control valves is two-position, marked as follows:

a. "automatic"

→ b. "close".

The remote manual (nonautomatic) switches are simple "open", "close" switches and are set as desired. They must be operator actuated to close the valve in case containment is necessary.

5.4 Reactor Coolant System

5.4.1 General Description

The reactor coolant system, Figures 5.4.1.1 and 5.4.1.2, uses helium to remove heat from the reactor core and to transfer this heat to water and steam in the steam generators. The reactor coolant is circulated through two coolant circuits, or loops, each containing one steam generator and one reactor coolant blower. The two reactor coolant blowers furnish a total flow of 430,000 lb/hr with a pressure rise of 16.75 psi at 100% flow conditions.

The helium enters the reactor vessel at approximately 508 F and 315 psia and leaves at 1043 F and 303 psia under full load conditions. The core design pressure differential is 10.25 psi. Table 5.4.1 shows the design temperature, operating temperature and pressure for each section of the reactor coolant piping.

TABLE 5.4.1

Operating Conditions at 100% Reactor Power

<u>Section</u>	<u>Operating Temp. (F)</u>	<u>Design Temp. (F)</u>	<u>Operating Pressure (psia)</u>
Reactor to attemperator outlet	1043	1200	303
Attemperator to steam generator	1043	1080	303
Steam generator to blower	483	600	298
Blower to reactor inlet	508	600	315
Bypass from blower to block valve	508	600	315
Bypass from block valve to attemperator	508	1080	303

5.4.2 Leakage Specification and Tests

→ The main coolant piping is shop fabricated in spool pieces incorporating piping sections, valves and fittings and is subsequently hydrostatically tested to the "after erection" requirements of the ASA Code for Pressure Piping B31.1, 1955, Paragraph 121.(3). Components of the remainder of the coolant system, valves, pipe sections and fittings, are hydrostatically tested as components according to the "before erection" requirements of this code. After erection, the system is pneumatically tested to 1.25 times the system design pressure. The pneumatic test is followed by an evacuation test designed to verify tightness of the system. The system is evacuated until the absolute pressure is fifty microns of mercury, or less, and the vacuum exhaust line is valved off. The pressure in the

TABLE 5.4.5.1 (continued)

Reactor Coolant Blower Design Data

Power input to gas at 100% load	1860 kw
Brake horsepower at 100% load	2542
Casing material	steel plate
Rotor material	4340 steel forging
Blade material	turbine quality stainless steel (403 modified)
Method of attaching blades	broached-dovetail
Bearing type and design life	sleeve - 20 years
Friction horsepower at 100% load	32 hp at 6750 rpm (estimate)
Running seal horsepower loss (shaft)	17 hp per seal (estimate)
WK ²	483 lb ft ²
Critical speed of rotor	8300 rpm

Tables 5.4.5.2 and 5.4.5.3 summarize the normal and depressurized performance characteristics of the reactor coolant blower. Figure 5.4.5.3 is a cross section view of the blower. ←

5.4.6 Blower Seal Leakage

Each reactor coolant blower is fitted with both static and running seals. The static seals hold helium leakage below an allowable rate when a blower is not running. The static seals consist of bellows welded to the inside of the blower housing concentric to and at opposite ends of the drive shaft. The static seals are actuated by helium at 390 psia which expands the bellows and seats their attached sealing rings against the rotor. →

The running seals are water-buffered floating ring type. An extensive program of testing was conducted to insure that these seals would provide a negligible rate of helium leakage to the atmosphere along the rotating blower shaft without introducing contaminants into the reactor coolant stream.

The buffering seal water flows into the seal cavity in a direction counter to the leaking helium. Mixing occurs in the seal cavity and the helium carried off in the liquid is recovered, purified and returned to the reactor coolant system. The seal water is continuously filtered before being recirculated through the blower seal cavity. ←

5.4.7 Piping and Expansion Joints

The reactor coolant piping from the steam generator outlet to the blower, from the blower to the reactor inlet, and the bypass line from the cold leg to the block valve in the bypass line, is fabricated of carbon steel per ASTM Specification A155, Class 1, Grade KC70, firebox quality, or ASTM Specification A212, Grade B plate, manufactured in compliance with ASTM Specification A300.

The design pressure of this piping is 350 psig at a design temperature of 600 F.

TABLE 5.4.5.2

REACTOR COOLANT BLOWER PERFORMANCE - NORMAL OPERATING CONDITIONS

		Percent Load Operation				
		110	100	74	60	20
Suction pressure of He	psia	296	298	297	295	288
Suction temperature of He	F	483	483	464	450	425
Mass flow per blower	lb/hr	236,000	215,000	159,000	129,000	43,000
Suction volume flow per blower	cfm	33,300	30,400	22,100	17,400	5,900
Pressure rise	psi	20.2	16.75	9.2	6.0	0.8
Blower bhp						
(Running seal loss not included)		3372	2525	1050	578	38.4
Blower flange to flange efficiency %		87	88	86	84	62
Blower fluid horsepower	hp	3340	2495	1021	556	33.4
Blower bearing losses ^a	hp	32	30	29	22	5
Blower seal losses	hp	33	30	16	11	1.6
Input horsepower to blower	hp	3405	2555	1066	589	40
Speed increaser losses	%	1.674	1.912	3.00	3.129	9.09
Input to speed increaser	hp	3463	2605	1099	608	44
Fixed losses in fluid coupling	hp	45	45	45	45	45
Slip losses in fluid coupling	hp	89	345	621	544	169
Motor power output	hp	3597	2995	1765	1197	258
Motor efficiency	%	93.3	93.3	91.6	88.9	64
Motor windage and friction losses	hp	90	90	90	90	90
Blower speed	rpm	6750	6130	4432	3620	1427
Motor speed	rpm	1780	1783	1790	1793	1798

^aBlower bearing losses include flywheel friction and windage losses.

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from document

central, dividing plate. A three-way valve on the tank inlet-outlet line, remotely operable from the main control room, permits the use of only one section of the tank at a time. Each tank section is fitted with a drain, an overflow, and a remote water level indicator in the main control room. A low water level alarm, common to both sections, annunciates in the main control room.

An 8-in. line from the deaerator supplies feedwater to feedwater pumps J-3-1 and J-3-2 and shutdown feedwater pumps J-6, J-7, J-64 and J-65 (Section 5.6.7). Feedwater pumps J-3-1 and J-3-2 are identical, horizontal, centrifugal, double casing type pumps, each rated at 610 gpm at approximately 1440 psi total head when pumping 287 F water. One pump normally supplies the total feedwater requirements of both steam generators, while the second pump is on stand-by. A continuous flow of hot feedwater through the stand-by pump is maintained by a warmup line which circulates water from the pump discharge, through an orifice and two globe valves, to the main condenser. The stand-by pump is automatically started on low feedwater flow in the suction line of the operating pump, and a low flow alarm is simultaneously actuated in the main control room. Each pump has a recirculation line to the deaerator which maintains a minimum flow of 70 gpm through the pump. The recirculation control valve can be operated in two modes; modulating, to maintain the total pump flow at 70 gpm, or two-position, set to open wide when flow through the pump falls to 70 gpm. The operator chooses the mode of operation by a selector switch. Under either mode, a low flow alarm is actuated in the main control room when the control valve opens. Each pump has local indication of lube oil pressure, bearing temperatures, suction and discharge pressure, and suction flow and temperature. Indication of pump suction flow and discharge pressure, and the low flow alarm are provided in the main control room. In addition, flow in the common 8-in. suction line to the feedwater pumps is recorded in the main control room. Both of the feedwater pumps are tripped by low deaerator water level (1000 gal).

The feedwater pump discharge lines combine into a common 6-in. discharge header which divides into two 4-in. lines outside the reactor building, each supplying one of the steam generators. Each 4-in. line has a feedwater regulating valve, operated by a three-element control, with a hand-operated regulating valve in parallel. Both of these valves are remotely operable from the main control room. The three-element control system controls the feedwater regulating valve by a steam generator drum water level signal which is modified by the ratio of steam flow to feedwater flow. Pneumatic or electric failure in the three-element control system causes the regulating valve to fail "as is". The hand-operated regulating valves can be used for initial filling and start-up, and during periods when the automatic valve is isolated for maintenance. Pneumatic failure of the controller causes the manually actuated valve to fail closed. A check valve and a motor-operated gate valve for containment isolation are located in each 4-in. line inside the containment building. An additional motor-operated gate valve is located outside the containment shell for isolation. Two check valves in series are provided upstream of the regulating valves. Piping ruptures upstream of these check valves can not affect the integrity of the water supply from the feedwater tank and the shutdown feedwater pumps which join the main feedwater line between the check valves and the regulating valves.

Pressure and temperature of the water in the 6-in. feedwater discharge header are recorded in the main control room. Flows and pressures in the 4-in. lines to the steam generators are recorded in the main control room; pressure and temperature are indicated locally.

5.6.6 Circulating Water System

Circulating water enters the upper half of the front water box of the main condenser through a 36-in. pipe. It passes to the rear water box through the

upper condenser tubes, reverses, and returns to the lower half of the front water box through the lower condenser tubes. Water leaves the condenser through a 36-in. pipe from the bottom of the front water box (Figure 5.6.6). A description of the circulating water system exterior to the turbine building is given in Section 6.11.2.

The priming ejector C-6, (Figure 5.6.2), removes air from the condenser tubeside and from the circulating water system during initial filling of the circulating water system. Steam is supplied to the priming ejector from the main steam line. The steam-air mixture is discharged to the blowdown tank F-42. Blowdown tank F-42 also receives the deaerator overflow dump, the steam generator drums continuous blowdowns, and miscellaneous steam generator drains. Water from the tank discharges to the floor drain system and steam from the tank is vented through a 12-in. pipe to the atmosphere (Figure 5.6.5.3). The ejector C-6 draws air from the top of the condenser water box inlet nozzle, and from the high points in the circulating water supply and return lines located just outside the turbine building. Sight tanks with gage glasses are installed in the vacuum lines from the circulating water line high points. These tanks are installed above the elevation of these high points so that the operator can observe when all of the air has been evacuated from the circulating water lines. A gage glass is installed on the condenser water box inlet nozzle so air evacuation from the condenser can also be observed. Priming during the initial filling of the circulating water system requires the manipulation of manually operated valves.

During normal operation, the condenser tubeside and portions of the circulating water piping are under a maximum vacuum of approximately 25 ft of water. A system for continuous automatic priming removes air from the water. Vacuum lines are installed from the top of the condenser rear water box, the top of the lower half of the condenser front water box, and from the circulating water line evacuation sight tanks referred to above. These vacuum lines are manifolded into two loop seals. A single 1-1/2-in. vacuum line runs from the outlet of the loop seals to air ejector C-5, which accomplishes the continuous priming. The loop seals prevent the drawing of circulating water (river water) into the air ejector and subsequently into the steam side of the condenser.

The circulating water temperature and pressure are indicated locally at the main condenser inlet and outlet. Circulating water pressure at the discharge of the circulating water pumps in the pump house is indicated in the main control room; a low pressure alarm is also installed in the main control room.

5.6.7 Shutdown Heat Removal System

a. General

The shutdown heat removal system removes the reactor decay heat following reactor shutdown. The system is designed to handle 5% of full reactor thermal power. The principal items of shutdown equipment are shutdown condenser C-14; shutdown drain cooler C-29; desuperheaters C-9 and C-12; desuperheater C-11; shutdown feedwater pumps J-6, J-7, J-64 and J-65; and shutdown make-up pumps J-19-1 and J-19-2. Additional items of equipment required for an emergency shutdown include feedwater surge tanks F-51 and F-56 and their nitrogen pressurizing systems and steam generator pressure blowdown valves PSV-1-145 and PSV-1-146.

During normal shutdown the operator can use the main condenser, main feedwater pumps and other main cycle equipment instead of the shutdown equipment for part of the shutdown period. Utilization of these components, however, will become increasingly inconvenient as reactor decay heat decreases.

Each make-up pump has a recirculation line with a globe valve and orifice for minimum flow. The two recirculation lines combine into a common line which discharges, through a three-way valve remotely controlled from the main control room, to one of the two sections of the condensate storage tank. Pump suction temperature and pressure and discharge pressure are locally indicated.

During shutdown operation, steam flow to the deaerator may be required to keep the deaerator pressure above atmospheric, depending upon the proportions in which cold make-up and hot condensate drains are supplied to the deaerator. Desuperheater C-11 supplies steam required to maintain a minimum pressure of 4 psig.

Each steam generator has its own independent system of shutdown feedwater pumps and piping. The system for each steam generator is identical, with the single exception that the suction piping for the shutdown feedwater pumps for steam generator C-1 is normally lined up so that the pumps take suction from the condensate storage tank, while the suction piping for the shutdown feedwater pumps for steam generator C-2 is normally lined up so that the pumps take their suction from the demineralized water storage tank. The following is a description of the shutdown feedwater system for steam generator C-1, only.

During shutdown operation, feedwater is supplied to steam generator C-1 by either shutdown feedwater pump J-6 or by shutdown feedwater pump J-7. Each pump has adequate capacity to supply total shutdown feedwater requirements. Pump J-7 is electrically driven and can be controlled both locally and from the main control center and can be locked out locally. Pump J-7 is arranged to start up automatically from any of the following: low deaerator level (1000 gal), loss of power accident, or low level in either steam drum (9 in. below normal level). After an automatic start, the pump can only be stopped manually (either locally or from the control room).

Pump J-6 is a horizontal duplex reciprocating steam driven pump and receives its 135 psig, 500 F steam supply from desuperheater C-12. The steam supply to the pump is controlled by a pump governor control valve PV-1-110, which throttles steam supply to the pump to maintain a constant pump discharge pressure. Pressure control valve PCV-1-111, of desuperheater C-12 automatically opens and provides 135 psig steam to the pump governor valve, and thereby starts pump J-6, from the same signals that start pump J-7. Pressure control valve PCV-1-111 can also be made to open and control manually from the control room. After automatic opening of PCV-1-111 it can only be closed manually. (Desuperheater C-12 and valve PCV-1-111 also supply steam, in the same manner, to steam pump J-64, for steam generator C-2.) The steam cylinders, and the piping from the desuperheater to the steam cylinders, are kept warm when the pumps are not operating. A small flow of steam is maintained within the piping by cracking the bypass valve around the desuperheater pressure control valve (Figure 5.6.1.2) and allowing steam to flow through the bypass lines around the pump and into the pump discharge lines. Testing of the motor and steam driven shutdown feedwater pumps is possible during normal plant operation.

Pumps J-6 and J-7 have several alternate sources of suction supply. The valves on a common suction header are normally left in position so that the pumps take their suction from the condensate storage tank on automatic start-up. The suction header feeds each pump through a separate check and gate valve. By manual manipulation of valves, supply to the suction header can be changed from the condensate storage tank to the

demineralized water storage tank. The supply to the suction header also has a tie from domestic water. On low pressure in the header, pressure control valve PCV-3-175 opens to supply domestic water to both pumps. Valve PCV-3-175 remains open until its pressure switch is manually reset. Valving is provided around valve PCV-3-175 to make possible periodic checks for leakage across a closed valve. The pumps are provided with a second suction header, feeding each pump through a separate gate and check valve, which draws water from the deaerator feedwater discharge line. This source of supply is used for normal shutdowns.

Each pump discharges through a check valve into a common line in which control valve HCV-3-28 is installed. HCV-3-28 is normally wide open, but can be modulated by hand controller HC-3-28 mounted in the main control room. The feedwater flow passes through two check valves and a globe valve, and into steam generator C-1 feedwater supply line, downstream of the double check valves on that line. Thus feedwater supply to the generator is safeguarded in the event of a line rupture anywhere in the main feedwater supply piping upstream of the 4-in. check valves. A direct supply from HCV-3-28 to the common 6-in. main feedwater supply line, with a normally closed valve, also is provided. This permits use of shutdown feedwater pumps J-6 and J-7 to feed both steam generators C-1 and C-2, if desired. (A similar cross-tie is provided from HCV-3-162, on the discharge of pumps J-64 and J-65, to permit use of these pumps to feed both steam generators, if desired.)

Each pump is installed with a pressure relief valve which protects the pump against over-pressuring which could be caused by check valve leakage. Since pump J-6 is a positive displacement pump, its relief valve also provides protection in the event the pump is started while the discharge valve is closed.

Each pump is provided with a minimum flow recirculation line. Each line is installed with a globe valve and an orifice (FX-3-5 and FX-3-11) for passing the minimum flow required by the pump.

In addition to supplying feedwater to steam generator C-1 during shutdown, pumps J-6 and J-7 supply water to desuperheaters C-9, C-11 and C-12. Control of desuperheating water flow is described in Section 5.6.3.

Separate discharge lines from each shutdown feedwater pump, each with a check and globe valve, furnish water to a common header which feeds the desuperheaters. The header is divided into three branches, each having a temperature control water valve, supplying desuperheaters C-9, C-12 and C-11, respectively. The header is constantly pressurized by a cross-tie line from the discharge of desuperheater water pumps J-8-1 and J-8-2.

c. Steam Generator Pressure Blowdown

→ In the event of a loss of electrical power, or on low steam drum water level resulting from loss of normal feedwater supply, the steam generator pressure must be blown down to 150 psia so that the shutdown feed-pumps can supply feedwater to remove reactor decay heat. To reduce steam pressure two pressure blowdown valves PSV-1-145 and PSV-1-146 are installed, one on each steam generator superheated steam outlet line. These valves open automatically on electrical power failure or on low water level in either steam drum and close when the steam pressure is reduced to 150 psia. Simultaneous with the opening of these valves the turbine stop valve is tripped through its solenoid device with a time

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RECLOSED BREAKERS PERFORM COMPLETE OPERATION IN 20 CYCLES. THE 900 AND 800 SERIES BREAKERS HAVE 3 TO 5 CYCLE OPENING TIME. THE 200 SERIES BREAKERS HAVE 6 CYCLE OPENING TIME.

FIG. 5.7.2.

practical purposes, therefore, the output of the EGCR electrical generator will not vary because of normal operating conditions on the power system to which the plant is connected.

On the other hand, if abnormal operating conditions occur on the external grid which cause low system electrical speeds, the underfrequency relay at the EGCR will trip breaker number S-1340 (13.8 kv line main breaker), leaving only the station service electrical load on the EGCR generator.

A high system electrical speed will cause the overfrequency relay at EGCR to trip breaker number S-1340, leaving only the station service electrical load on the EGCR generator.

Relaying systems protect the EGCR from faults which occur on electrical equipment and lines in the external grid. Three areas in the external grid are considered: EGCR to X-10 line equipment protection, X-10 substation protection, and protection of 161 kv lines, as it affects the EGCR.

The EGCR to X-10 line equipment has the following protection at the EGCR (breaker S-1340):

- a. Pilot wire differential current relaying (in connection with X-10 relaying)
- b. Directional overcurrent relaying for ground fault protection, current polarized
- c. Instantaneous and time overcurrent relaying
- d. Time overcurrent relaying for transformer neutral ground faults
- e. Transformer sudden gas pressure relaying
- f. Transformer load-tap-changer protective devices
- g. Remote tripping from a "b" finger of X-10 breaker number 264.

Additional relays in the EGCR 13.8 kv bus operate breaker S-1340.

The EGCR to X-10 line equipment has the following protection at X-10 (breaker 264):

- a. Pilot wire differential current relaying (in connection with the EGCR relaying)
- b. Instantaneous and time overcurrent relaying
- c. Bus differential relaying (on the X-10 bus)
- d. Impedance relays coordinated with the carrier pilot relaying of the 161 kv lines (All ground faults and most of the phase faults on the 161 kv lines will cause opening of breaker 264.)
- e. Power directional (reverse current) relays in the circuits to breakers 874 and 884 will cause breaker 264 to trip on power flow from the EGCR to the indoor 13.8 kv bus at X-10 or to the 161 kv power lines at X-10 if, at the same time, there is low voltage on the power lines. This prevents electrically overloading the EGCR generator during emergencies
- f. High speed reclosing of breaker 264 (one shot).

To summarize the X-10 - EGCR line protection, any fault between X-10 and EGCR 13.8 kv bus opens breaker S-1340 at the EGCR and breaker 264 at X-10. If the fault clears immediately, breaker 264 recloses. Breaker S-1340 can be reclosed only after breaker 264 recloses, and then only by the operator who must re-synchronize and manually close breaker S-1340 from the EGCR control room. If breaker 264 locks out for any reason, it must be closed by local manual operation or by remote manual operation (via supervisory control) from K-25.

In addition to the protection with breaker 264 at X-10, the relaying which affects the EGCR under normal operation includes protection for the 40,000 kva transformer and its associated buses at the X-10 substation. (Breaker 204

normally is open and, therefore, the 35,000 kva transformer and its associated buses are not involved in the EGCR relaying.) This relay protection consists of:

- a. Transformer and bus differential relaying
- b. Transformer sudden gas pressure relaying
- c. Time and instantaneous overcurrent relaying on the bus
- d. Time overcurrent protection against ground faults (This is secondary or backup protection which functions when instantaneous overcurrent and directional overcurrent relays fail to function properly.)
- e. Directional overcurrent protection against ground faults
- f. Thermal overload protection for the transformer.

The above relaying trips both 161 kv breakers 884 and 894 and the 13.8 kv breaker 284 for faults between these breakers. The two breakers also operate for faults on the EGCR, ORIC and HFIR 13.8 kv lines if the respective breakers; 264, 254 and/or 294; malfunction. ←

The basic 161 kv protection for the K-25 - X-10 and the X-10 - Elza lines consists of the following: ←

- a. Carrier pilot relaying for phase-to-phase and phase-to-ground faults
- b. Impedance type (backup protection) distance relaying
- c. Ground fault relaying (backup protection)
- d. High speed reclosing (one shot).

The above relay schemes open (and reclose one shot after a fault) breakers 904 at Elza station and breakers 884 and 894 at X-10 or breakers 924 and 926 at K-25 station and breakers 874 and 894 at X-10. One transmission line at K-25 and three transmission lines at Elza station are the major elements of the local electrical grid. The relay protection associated with each of these transmission lines is very similar in every respect to that described for the Elza - X-10 and K-25 - X-10 transmission lines. This protection makes the service reliable so that, except for instances of extreme system troubles on the external power network, the Elza and/or the K-25 161 kv buses will be energized at all times. Major transmission troubles that may interrupt service to and from the EGCR are expected to come from first, the 13.8 kv system and second, the 161 kv circuits from the K-25 or Elza stations. The EGCR-X-10 circuit is a 1595 MCM aluminum line with two conductors per phase. It is constructed on a wooden H frame equipped with two overhead ground wires, and the entire construction was designed for 100 mph wind. The line is well protected against normal hazards. ←

5.7.3 Plant Generator Facilities

The plant electrical generation facilities, as shown in Figures 5.7.3.1 and 5.7.3.2, consists of the 31,250 kva generator and the 13.8 kv generator bus switchgear, which includes circuit breakers for the main isolating transformer and transmission line, the two reactor coolant blower motors, the two auxiliary power transformers and two feeders to the experimenters' power transformers. The generator is a 13,800 v, three phase, 60 cycle, 3600 rpm, hydrogen-cooled unit, nominally rated 25,000 kw, 0.8 pf, 31,250 kva, 0.9 short circuit ratio, with 0.5 psig hydrogen pressure. Provision is made for operating at increased hydrogen pressures up to 15.0 psig and a corresponding increase in generator capacity up to 35,927 kva (115%).

Excitation for the generator field is supplied from a motor generator set consisting of a 125 kw, 250 v, d-c generator with a 3 kw, 250 v, d-c pilot exciter and a 400 hp, 1189 rpm, 2400 v induction motor. A motor operated, rheostatic-type voltage regulator, electrically operated field circuit breaker with field

because of failure of local protective devices. These relays, except the reverse power relay, trip the main generator and field circuit breakers together with the turbine stop valve only after adequate time has been allowed to permit operation of primary protective devices. The reverse power relay trips the main generator breaker only. The negative phase sequence relays which measure current unbalance in the generator phases are equipped with separately timed alarm contacts which close in advance of the tripping contacts. The amount of advance warning depends on the magnitude of unbalance, and in case of minor unbalances, will allow sufficient time for the operator to endeavor to take corrective action. ←

The effects of tripping the generator by these backup relays are the same as for the generator protective relays, with the added possibility that in some cases the backup relays on the transmission circuit may also operate and interrupt all power to the plant. In such a case, the emergency generators automatically start and supply emergency power. ←

If the external power supply for the transmission system is lost while the generator and transmission system are operating in parallel, and the turbine is operating on initial pressure regulation, the generator will probably become overloaded or underloaded with a resulting change in speed and frequency. An underfrequency or overfrequency relay disconnects the plant generation system from the transmission system in such cases by tripping the circuit breaker for the main isolating transformer and transmission line. The relays also trip the initial pressure regulator and return the turbine-generator to governor control, thus restoring normal speed and frequency. The turbine-generator continues to supply power for operation of the reactor and plant auxiliaries.

A scram signal from the reactor safety system trips the turbine stop valve which in turn trips the generator main circuit breakers to disconnect the generator from the system. ←

The 13.8 kv generator bus switchgear consists of 10 units of standard metalclad switchgear with a single section 2000 amp bus and eight air-break, metalclad circuit breakers having an interrupting rating of 500 Mva, 21,000 rms amp at 13.8 kv and a momentary rating of 40,000 rms amp. The circuit breakers for the generator and for the transmission line isolating transformer are rated 2000 amp continuous. All other circuit breakers are rated 1200 amp continuous. Two units of the switchgear are auxiliary compartments for potential transformers.

The switchgear bus is equipped with bus differential relays. The zone of protection provided by these relays includes all eight circuit breakers connected to the bus. In case of a fault on the bus or in a circuit breaker, the relays trip all of the circuit breakers. Tripping of all circuit breakers interrupts the power supply to reactor and plant auxiliaries and causes the emergency diesel engine generators to start to supply emergency power.

The circuit breaker for the main isolating transformer and transmission line is connected to the transformer by 2000 amp metalclad bus. This bus is included in the zone of protection provided by the protective relays for the transformer and transmission line. Control devices for the circuit breaker and the isolating transformer tap changer are located on the electrical control desk in the main control room.

Indicating instruments, tap-changer position indicator, recording meters, protective relays, pilot wire relays and alarm annunciators for the isolating transformer and transmission line are located on the duplex switchboard in front of the control desk.

The two circuit breakers for the plant auxiliary power transformers are equipped with primary and backup protective relays and are connected by 15 kv armored

cable to the primaries of the transformers. These cables are included in the zones of protection provided by the differential relays for the transformers. Further discussion of these circuits is contained in Section 5.7.4. Control devices for these circuits are located on the electrical control desk in the main control room. Indicating instruments, protective relays and alarm annunciators are located on the duplex switchboard in front of the control desk.

Two circuit breakers are installed for the feeders to future experimenters' auxiliary power transformers. Controls, instruments, relays and alarms are identical to those for the plant auxiliary power transformers and are located on the electrical control desk and duplex switchboard in the main control room.

The two circuit breakers for the reactor coolant blower motors are equipped with primary and backup protective relays and are connected by 15 kv armored cable to the motors. The cables are included in the zones of protection provided by the differential relays for the motors. Undervoltage relays disconnect the motors in case of loss of power on the 13.8 kv switchgear bus. Voltage and current surge protection equipment is installed at the motor terminals. Control devices, indicating ammeters and alarm annunciators for these motor circuits are located on the helium switchramp in the main control room and local controls are installed at the motors. Protective relays and indicating ammeters for these circuits are located on their respective metalclad switchgear units. The control circuits for these motors are integrated into the reactor instrumentation and control circuits.

5.7.4 Station Auxiliary Power System

The auxiliary power system, shown on Figure 5.7.3.1, consists of the dual 2.4 kv and 480 v systems which furnish the normal power to the plant auxiliaries.

The 2.4 kv auxiliary power system is shown in more detail on Figure 5.7.4.1. It consists of two 5000 kva, 13.8-2.4 kv auxiliary power transformers, the two-section 2.4 kv switchgear, and associated transformer feeders and motor circuits.

The primaries of the two 2.4 kv auxiliary transformers are supplied from circuit breakers in the 13.8 kv switchgear as described in Section 5.7.3. The secondary of each transformer is connected by a 2000 amp metalclad bus through a 2000 amp circuit breaker in the 2.4 kv switchgear to one section of the 2000 amp bus in the switchgear. The transformer secondary metalclad buses are included in the zones of protection provided by the differential relays for the transformers. The secondary neutral of each transformer is grounded through a grounding resistor rated 600 amp for ten seconds. Each transformer is equipped with a gas pressure relay for fault detection, in addition to differential relays and backup phase and ground fault relays. Operation of any of these relays trips the 13.8 kv primary breaker and the 2.4 kv secondary breaker for the transformer in difficulty.

The 2.4 kv auxiliary power switchgear consists of 14 units of standard metalclad switchgear with a two-section 2000 amp bus, a 1200 amp bus tie circuit breaker, four transformer feeder circuit breakers, one fire pump motor feeder circuit breaker, and five motor starting circuit breakers in addition to the two circuit breakers for the incoming secondaries of the auxiliary power transformers. One unit of the switchgear is for the bus transition for the bus tie breaker. All circuit breakers have an interrupting rating of 150 Mva, 36,100 amp at 2.4 kv and a momentary rating of 37,500 amp. All transformer feeder and motor circuit breakers have a continuous rating of 1200 amp.

Each of the two sections of switchgear bus is equipped with bus differential

operated and is key interlocked with a similar breaker in the emergency power load-center. These two breakers feed the second lighting feeder distribution panel which supplies the remaining half of the plant lighting. The key interlock prevents closing both feeder circuit breakers at the same time. The load can be transferred from the normal load-center to the emergency load-center by tripping the normal breaker and transferring the key to the emergency breaker to permit it to be closed. The same procedure is required to transfer the load back to the normal load-center.

The six circuit breakers for the motor control center feeders are manually operated. No interlocks or special controls are required for these feeder circuit breakers. These six feeders supply normal auxiliary power to 480 v motor control centers 4, 5, 6 and 7. These motor control centers supply normal power for non-critical plant auxiliaries and services which are not required for an emergency shutdown or scram of the reactor.

The motor control centers are unitized metalclad structures with 600 amp main buses, incoming main circuit breakers, combination type motor starters, feeder circuit breakers, control transformers and control devices.

Motor control centers 4 and 5 supply plant process auxiliaries and have single main bus sections with two incoming mains with large air circuit breakers. The incoming mains are supplied from opposite sections of the normal power load-center and the circuit breakers are key interlocked to prevent parallel operation of the main feeders. The circuit breakers are manually operated and, in case of loss of the operating feeder or load-center section from which it is supplied, will require manual transfer to restore power to the control center bus.

Motor control centers 6 and 7 supply building services and have single main bus sections with single main incoming mains with molded case circuit breakers.

5.7.5 Emergency Power System

The plant emergency power system consists of a dual 2.4 kv - 480 v load-center substation, two emergency diesel engine-driven generators, three 2-section motor control centers, power feeders, branch circuits, motor circuits and control circuits to supply either normal or emergency power to essential plant auxiliaries.

The dual 2.4 kv - 480 v load-center is shown on Figure 5.7.5.1 and consists of two single-ended load-centers, each with its own transformer and metalclad switchgear, arranged end to end to form an integrated unit with copper bus bar connections between all components of each section of the load-center. There is no bus tie between the two switchgear sections.

Each of the two emergency power load-center sections consists of a 862 kva, 2.4 kv - 480 v askarel insulated transformer throat-connected to the 480 v metalclad switchgear. The neutrals of the transformers are solidly grounded. Each transformer has a separate 5 kv armored cable primary feeder fed from separate sections of the 2.4 kv switchgear described in Section 5.7.4. The transformer secondaries are connected through 1200 amp air circuit breakers to their respective switchgear buses.

The transformer secondary breakers are equipped with integral short-circuit trip devices and overcurrent relays for coordinated tripping with the feeder circuit breakers on a selective trip basis. A fault on a feeder is cleared by the feeder circuit breaker without tripping the main circuit breaker. The main circuit breaker trips for a fault on the main bus or if a feeder breaker fails to clear a fault on the feeder. The main circuit breakers are

interlocked with their respective 2.4 kv transformer primary circuit breakers so that tripping of the 2.4 kv breaker trips the corresponding 480 v secondary main breaker.

Control switches for the secondary main breakers are located on the duplex switchboard in the main control room. Annunciators are located on the duplex switchboard to indicate loss of power on either 480 v switchgear bus. Ammeters and protective relays for the secondary mains and bus voltmeters are located on the 480 v switchgear.

The two groups of 480 v switchgear for the emergency load-center are identical and, in addition to the transformer secondary main circuit breakers, each group includes two 2000 amp main circuit breakers for the two emergency generators, five feeder and three motor starting circuit breakers as follows:

- a. Emergency motor control center 1
- b. Emergency motor control center 2
- c. Emergency motor control center 3
- d. Failure-free power control panel
- e. Emergency feeder for lights
- f. Blower seal water pump motor
- g. Vessel cooling compressor motor
- h. Emergency cooling loop compressor motor.

All feeder and motor starting circuit breakers are equipped with integral over-current and short-circuit trip devices and are coordinated with the main breakers on the selective trip basis.

The feeder circuit breakers for the emergency motor control centers and the failure-free power panel are manually operated. No interlocks or special controls are required for these feeder circuit breakers since all of these breakers are normally closed to provide dual feeders to the control centers and the failure-free power panel. In the event of loss of power supply, either normal or emergency, to either section of the emergency load-center, the buses of the emergency motor control centers which were being supplied from that section of the load-center are automatically transferred at the control centers to the feeders from the section of the load-center remaining in operation. The feeders to the failure-free power control panel require manual transfer at the failure-free control panel as noted in Section 5.7.6. After power is restored to the load-center bus section, the motor control center buses may be transferred back to their normal condition by manual controls at the motor control centers.

The lighting feeder circuit breaker connected to switchgear bus A is manually operated and is key interlocked with a similar circuit breaker in the normal power load-center to permit manual transfer of one-half of the plant lighting load from the normal load-center to the emergency load-center as described in Sections 5.7.4 and 5.7.7.

The lighting feeder circuit breaker connected to switchgear bus B is electrically operated and is arranged in an automatic transfer scheme with a similar circuit breaker in the normal power load-center to automatically transfer one-half of the plant lighting load from the normal load-center to the emergency load-center as described in Sections 5.7.4 and 5.7.7.

The circuit breakers for the two vessel cooling compressor motors are electrically operated and are arranged to operate with either compressor running under manual control and the other compressor on stand-by and arranged to start automatically by a pressure differential signal in case of failure of the running compressor. In the event of loss of normal power on both of the emergency load-center buses, the stand-by compressor will start automatically if normal

power is restored, or when emergency power is supplied to the switchgear bus to which it is connected. A timing relay delays the automatic restart for 30 sec after the bus is energized to delay the application of this motor-starting load to the generator until the generator voltage and frequency have recovered from the effects of the initial application of load. If both engine-generators start and function properly, the total outage time for the compressor is approximately 42 sec. If one engine-generator fails to start or if automatic switching operations are not performed properly, the outage time for the compressor is either 42 or 59 sec, depending on which compressor has been pre-selected as stand-by and which generator fails. The control switches for the motors are located on the helium control switch ramp in the main control room.

The circuit breakers for the two blower seal water pumps are electrically operated and are arranged for both pumps to be running normally under manual control. Following the loss of normal power on both of the emergency load-center buses, one pump restarts automatically if normal power is restored, or when emergency power is supplied to the switchgear bus to which it is connected. A timing relay delays the automatic restart after the bus is energized to delay the application of this motor-starting load to the generator until the generator voltage and frequency have recovered from the effects of the initial application of load. Each pump has a timing relay, one set for 5 sec and the other set for 7 sec, to prevent starting both pumps if power is restored to both buses simultaneously. If both engine-generators start and function properly, the total outage time for the pump is approximately 17 sec. If one engine-generator fails to start or if automatic switching operations are not performed properly, the outage time for the pump is either 17 or 19 sec, depending on which generator fails. If both engine-generators start properly, the second seal water pump may be restarted manually. The control switches for the motors are located on the helium control switchramp in the main control room.

→ The circuit breakers for the two emergency cooling loop compressors are electrically operated. Either compressor may be selected for operation with the second unit in standby. Either compressor may be manually started, stopped or alternated from the auxiliary control room adjacent to the main control center. Power service is provided from normal plant power or via the diesel engine
→ driven generators.

The two emergency diesel engine-driven generators are each rated 0.8 pf, 1500 kva, 720 rpm, 480 v, 3 phase, 60 cycle, and are each equipped with a belt-driven 2140 rpm, 125 v, d-c exciter. Each generator can supply the emergency load for the essential plant auxiliaries. Capacity is also available to supply one-half of the plant normal lighting load. Both engine-generator units are arranged to start automatically in the event of loss of the normal power supply but if one unit fails to start or perform properly, the other unit will pick up all emergency loads by means of automatic switching operations in the emergency load-center and in the emergency motor control centers.

Each diesel is governed to maintain a speed of 720 rpm by an isochronous enclosed hydraulic type Woodward governor.

The diesel engines and their accessories are arranged as shown on Figure 5.7.5.2. The diesels are General Motors Model No. 16-278A, 2 cycle, 16 cylinder, 8-3/4-in. bore x 10-1/2-in. stroke.

Each diesel is equipped with a shaft driven air blower, cooling water pump, fuel oil pump, and lube oil pump. The exhaust gases are discharged through silencers, and an exhaust pipe which leads to the turbine building roof.

Each diesel has a fuel oil day tank of 275 gal capacity, which is automatically refilled from an outdoor underground storage tank on low day tank level by

automatically-operating transfer pumps and transfer line valves. The shaft driven, positive displacement fuel oil pump takes suction from the day tank and delivers fuel oil, through a strainer and filter to the supply manifold which fills the cylinder fuel injectors. A stand-by, motor driven fuel oil pump is installed in parallel with the shaft driven pump. The stand-by pump automatically starts and delivers fuel oil on low fuel oil pressure.

The shaft driven, positive displacement, lube oil pump draws oil from the

Pages 5-85 thru 5-106 are missing from document

operate at the ambient conditions in the charge machine room. Only those valves in contact with helium are discussed in this section.

The valve packings used for helium service on the charge machine are of two types: bellows sealed packing for valves smaller than 4-in., and double packed stem with a lantern ring leak-off which limits leakage to the atmosphere. The leak-off from the double packed valves is piped to the low pressure helium recovery system. The body and bonnet materials are either forged or rolled steel in compliance with the specified applicable codes and standards.

All valves have metal-to-metal seating surfaces and back-seating provisions. End connections are welded.

For bellows sealed valves, the total permissible in-leakage with a fully extended bellows is 0.01 micron cfh, when testing valves in accordance with MIL-STD-271A(6) and seat leakage is limited to 20 micron cfh. Double packed valves, when tested at design pressure and in an unseated position, have a total permissible leakage to the leak-off region of 0.1 scfh and leakage to the atmosphere of 0.01 scfh. Total seat leakage does not exceed 3 scfh.

A sample valve of each valve number and type is subjected to cyclical life testing at design pressure and temperature to ascertain if leakage rates and bellows design are adequate for the valves selected.

5.9.9 Leak Tests

The charge machine pressure vessel is shop tested for leakage upon completion of shop assembly, stress relief, and hydrostatic testing. The specified helium leak testing procedures for the shop are in accordance with MIL-STD-271A(6) or an air in-leakage sensitivity test. For the helium leak test the gross allowable leakage is 0.1 standard cm³/sec and the air in-leakage does not allow a pressure increase in excess of 3 microns/hr over a four-hour test period.

The completely assembled charge machine is again leak tested in the field after erection. This test is identical to the air in-leakage shop test; but includes in addition to a retest of the vessel, the sealed penetrations (electrical leads, piping, etc.), drive housing blisters, nozzles and valves (except the ball valve), manways and all other pressure containing units which are structurally integral to the pressure vessel. The field test is unsuited to the leak testing of two charge machine components; the ball valve and the attachment bellows. These components are tested separately.

The attachment bellows are pressurized to 525 psig, immersed in alcohol, and checked for gas bubble evolution for a five-minute period. Absence of bubble formation during the test is accepted as proof of leaktightness. The ball valve assembly is tested at full operating pressure and temperature for body and seat leakage. The maximum ball valve leakage which is allowed is 0.01 scfh.

The above tests assure that the total helium leakage from the charge machine is negligible.

5.9.10 Decontamination Provisions

Decontamination of the charge machine is not expected to be required more often than once a year and then only for maintenance and replacement of machine internal mechanisms. Gross or general decontamination will take place in the charge machine room.

If a fuel element failure occurs within the charge machine either by pinhole failure or mechanical damage, the machine is decontaminated prior to any repairs or preventive maintenance. The decontamination process reduces the activity level of the internal components to that point at which they may be safely removed to the reactor service building for additional decontamination and maintenance. The decontamination provisions in the machine reduce activity to acceptable levels.

The decontamination equipment comprises a series of sprays which completely wet the internals of the machine, the solution tanks which mix and heat the cleaning fluids, the pumps, piping and necessary miscellaneous fittings. The pumps, tanks, and miscellaneous fittings which serve the charge machine decontamination area are located in the reactor service building and are shown in Figure 5.9.10.

Eighteen spray nozzles, each fitted with three spray heads rated at 14 gpm at 125 psig, introduce decontamination fluids into the vessel and distribute the fluids over the internal mechanisms. The spray heads are the full-cone type with a 91 deg apex angle arranged for overlapping spray patterns. Each spray nozzle connection is normally fitted with an internal shield plug to complete the continuity of the external shielding.

Two steel tanks of 2500 and 4000 gal capacity, designed to operate at atmospheric pressure and 150 F temperature, are located in the reactor service building. Each tank is equipped with a motor-driven mixer and heating coils to condition the solutions prior to their injection into the spray system. These facilities are described in Section 6.7.

The principle decontamination reagent used in the system will be peroxide-inhibited solution developed by ORNL. This is an aqueous solution of 0.4M ammonium oxalate, 0.1M citric acid, and 1.0M hydrogen peroxide at pH 2.5 and 140 F. The solution is capable of dissolving uranium dioxide, gives good decontamination of fission products, and has low corrosivity of carbon steel (less than 0.001 mil/hr).

The decontamination procedures for the charge machine are discussed in Volume II of the Hazards Summary Report.

5.10 Reactor Service Machine

5.10.1 General

The reactor service machine operates within a limited access area located above the reactor vessel. The general arrangement is shown in Figure 5.10.1.

The mechanisms within the service machine pressure vessel are controlled remotely from the service machine control console which is located in the service and charge machine control room. The service machine permits access to the top of the core through a reactor top nozzle.

Reactor pressure vessel penetrations are described in Section 4.5. Each control rod nozzle permits access to as many as 16 fuel channels (depending on location) as well as the control rod channel over which it is located. Each special plug nozzle permits direct vertical access to a fuel channel. Each experiment nozzle permits access to core locations reserved for future experimental use. The service machine tools are designed to engage with the breech type closures installed in the top nozzles.

Each reactor top nozzle flange is equipped with a backup blind flange designed to contain reactor pressure. The space between the blind flange and the closure is continuously monitored for leakage during reactor operation. Manual operation is required to remove the blind flange and to connect or disconnect the service machine to or from a reactor vessel top nozzle.

The service machine is designed to perform the following functions during full power operation, except as noted:

- a. Insert and withdraw control rod drive mechanisms, shrouds, shock absorbers, and control rods for maintenance or access to the reactor
- b. Load and unload reactor fuel (at or below 20% reactor power and flow)
- c. Remove and install top nozzle dummy shield plugs
- d. Install and remove temperature instrumentation for calibrating the permanent fuel channel outlet gas thermocouples and the pneumatic temperature monitoring system.

→ The service machine is designed to perform the following functions while the reactor is shut down:

- a. Remove and install an instrumented fuel column (at or below 20% flow)
- b. Remove and install neutron sources
- c. Install remote viewing equipment or remote reactor and core measuring equipment (depressurized to 5 psig or below)
- d. Remove and install experiments (This function may also be performed at full or reduced power depending on the type of experiment)
- e. Retrieve control rods which may have fallen into the reactor core as a result of control rod cable failure
- f. Work jointly with the charge machine in dislodging fuel assemblies jammed in a reactor core fuel channel
- g. Remove bottom dummy (flow at or below 20%).

5.10.2 Service Machine Description

a. Pressure Vessel

The service machine pressure vessel is an upright cylindrical carbon steel vessel approximately 71 in. ID and 52 ft high. All materials stressed by pressure are ultrasonically and Charpy impact tested.

SERVICE MACHINE

FIG 5 10 1

Welds are x-rayed and tested by magnetic particle or dye-penetrant or non-destructive methods during fabrication. After hydrostatic test the vessel is helium leak tested.

→ The vessel is designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII, 1959 Edition, including nuclear code case interpretations(1). All carbon steel plate from which the shell and nozzles are fabricated to ASTM A-212 Grade B Fire-box quality and ASTM A-300 Specifications. The forgings for the flanges are fabricated with materials to ASTM A-350LF1, (70,000 psi minimum tensile stress).

The vessel is constructed in two flanged sections which are bolted together at final assembly. The vessel is designed with flat top and bottom head closures.

The principal vessel penetrations include the drive blisters, manholes and the 12-in. bottom nozzle. Pressure tight electrical connections are flanged to the top and bottom heads and to the drive blisters. Helium leakage at all flanges is minimized by double gasketed joints with helium leakoff connections between the gaskets. This space is monitored and vented to the recovery system.

b. Closure Valve

A 12-in. diameter sliding disc valve is bolted to the inside of the bottom flange of the vessel. It is designed to seal the interior of the service machine pressure vessel from the service machine room. The valve can effectively seal from either direction.

c. Expansion Joint

The service machine to reactor vessel connection is effected by a 12-in. diameter flanged expansion joint which is bolted to the vessel bottom head (Figure 5.10.2.1). The joint is designed to compensate for differential thermal movement between the reactor top nozzle and the service machine. In addition it is designed to minimize the pressure thrust loads imposed on the reactor vessel nozzles, by controlling the pressure between the two gas seals. Gas leakage past the upper seal to a pressure control valve downstream of the bleed line maintains a back pressure and controls the pressure thrust exerted by the joint.

The lower end of the expansion joint is flanged to match the flange on the reactor vessel top nozzles. This flange is manually connected to a reactor vessel top nozzle prior to service machine operations. Raising and lowering of the flange to and from a reactor nozzle is accomplished by actuating pneumatic cylinders.

d. Service Machine Crane

→ The pressure vessel is mounted on a movable bridge and trolley crane which accurately positions the service machine over reactor vessel top nozzles, storage holes, spent fuel transfer shaft, and the fuel loading and rehearsal areas. Two-speed bridge and trolley drive motors permit coarse and fine positioning of the service machine. Alignment of the service machine and reactor vessel nozzles is controlled from local pendants which position the service machine horizontally and which also permit vertical adjustment. To prevent movement of the machine while it is connected to a reactor vessel nozzle, bridge and trolley track shear locks are applied by manual operation of the track lock selector

SERVICE MACHINE EXPANSION JOINT ASSEMBLY

FIG 51021

→ switches on the pendant control. To insure that the crane drive units cannot be energized during reactor servicing, electrical interlocks prohibit energizing the drive units whenever the service machine expansion joint is not fully retracted.

e. Tools and Mechanisms

The service machine tools and mechanisms are mounted on a turret which rotates within the service machine pressure vessel. Indexing of the turret positions the required mechanism or tool above the service machine outlet nozzle. After a tool is selected and indexed over the nozzle, the turret is locked in position. The drive which locks the turret also unlocks the service machine closure valve. Figure 5.10.2.2 shows the arrangement of service machine tools on the turret.

1. Reactor Nozzle Closure Tool

This tool positions the breech lock segments of the nozzle closure within a reactor top nozzle and compresses the closure gasket by rotating the closure nut. Reversing the operation of the tool releases the gasket and positions the breech for removal of a top nozzle shield plug.

2. Plug Removal Tool

This tool removes top nozzle shield plugs from the reactor after the closure gasket has been unseated and the closure breech ring positioned for withdrawal by the nozzle closure tool. The tool consists of a grapple suspended from a crosshead supported on two vertical ball screws. The grapple paws engage the breech plug and the crosshead is raised to withdraw the grappled plug into the service machine. This tool can also replace shield plugs.

3. Plug Replacement Tool

This mechanism is identical to the plug removal tool and is used to install a new or reconditioned shield plug into a reactor top nozzle.

4. Fuel Chute

The fuel chute is designed to guide the fuel assembly grapple into a core fuel channel to remove or install fuel assemblies. The fuel chute operates in conjunction with the fuel hoist and grapple, and guides the fuel grapple into a selected fuel channel. Access to the reactor core is made through a control rod nozzle after a control rod drive or shield plug has been removed from the nozzle. The chute is a long tubular mechanism which is indexed and positioned over reactor core fuel channels by remote control. The lower portion of the chute contains a basket which can be radially positioned over any of the sixteen fuel channels surrounding a control rod channel. The fuel chute is also used to position other special components such as the temperature calibration instrument and the neutron source grapple.

5. Fuel Hoist and Grapple

The fuel hoist removes or installs fuel assemblies through the fuel chute and into or from the reactor core fuel channels. New fuel is stored in tubes within the machine. Spent fuel is stored in an

motor and each clutch is wired to independent electrical circuits. Failure of one motor will not render the drive inoperative. Disengagement of the clutch downstream of a defective motor and energizing of the alternate control circuit permits the use of the alternate electric motor. Each clutch is energized through duplicate d-c power circuits and the excitation of the coils is controlled by a variable rheostat adjusted to give the torque required to operate the specific mechanism. The torque limiting capability of the clutch circuits precludes damage to the mechanisms.

In addition to the rotation required of the drives, each drive blister, other than blisters 4 and 8, is designed to extend and retract the drive shaft from the blister to engage or disengage a gear coupling of a mechanism mounted on the turret. The shaft extension mechanisms also have duplicate motors and clutches to maintain power to the mechanisms.

In the unlikely event of failure of both motors, local manual operators are available for use at each drive blister. However, the reactor must be shut down to permit personnel access into the service machine room.

All drive mechanisms are fully enclosed in a gear box independent of the pressure vessel cap and are lubricated by forced splash lubrication. Lubricants compatible with the environment have been selected to minimize contamination of the reactor coolant gas.

Each drive motor is electrically interlocked to prohibit movement of a component which may damage any other component.

g. New Fuel and Spent Fuel Storage

New fuel is stored within the service machine in two vertical carbon steel tubes. There is adequate storage space for six fuel assemblies and one top dummy assembly. The spent fuel tube consists of two concentric carbon steel pipes with shielding lead poured in the 1-1/2-in. annulus. This shielding attenuates the gamma radiation from the spent fuel.

h. Electrical System

All control and power circuits which operate the mechanisms of the service machine are duplicated. A failure of one limit switch, motor, or other electrical device is automatically backed up with a duplicate electrical component. Each circuit is individually fused. All critical positions of components are indicated on the control console. A system of alarms, indicating lights, and instrumentation, and other methods of indicating the condition of the machine have been included in the design. A system of interlocks prohibit the operator from damaging the machine by improper sequence of operation. In addition, the mechanisms have been designed to include mechanical stops which, in conjunction with the torque limiting clutches, will stall the mechanisms before damage can occur. All motors are equipped with current overload trips.

i. Process Instrumentation

Process piping and instrumentation permits evacuation and pressurization of the service machine, and controls pressure and flow of helium coolant to the service machine. All operations for evacuation, pressurization and depressurization are manually performed in the service machine room except for depressurization of the expansion joint. A pressure

additional shielding. Coolant is introduced through a tube which penetrates the lead plug in the bottom of the cask.

The 12-ft high shielding for the instrumented fuel column is in the form of a triangular cross sectional shell. The shell houses the ball screws and permits passage of the hoist crosshead. Wall thickness is comprised of 2 in. of lead and 7/8 in. of steel.

Under normal conditions the control rod shroud is stored in the lower portion of the machine; however, in the event of broken control rod cable, it is necessary to raise the shroud into the upper portion of the machine. This condition contributes a maximum of 7.9 mrem/hr to the dose rate at the floor level. Portions of the surface of the upper shield will show a maximum of 1.2 rem/hr. Access to this region however, will be required only in the event of both operating and spare drive motor failures.

5.10.5 Service Machine Heat Removal

During operation, helium is supplied to the service machine at a pressure slightly in excess of reactor vessel upper plenum pressure. The helium flow is thus directed from the service machine downwards into the reactor vessel, and maintains the service machine and internal mechanisms at a maximum temperature of 150 F. A portion of the helium supplied to the service machine is diverted through the internal spent fuel shield when spent fuel is stored within the shield. The helium coolant to the service machine is supplied from the vessel cooling system. The gas is cooled to 125 F before being directed into the service machine. During the turret operation, the service machine closure valve is shut and the coolant gas is bypassed around the valve into the reactor nozzle.

5.10.6 Service Machine Decontamination

The service machine is fitted with permanently installed internal headers and spray nozzles through which decontaminating fluids can be sprayed throughout the machine. Should the machine become contaminated, it is first thoroughly rinsed with detergents and water and re-checked for radiation levels. If high levels are noted, a chemical decontamination cycle is performed and is repeated until acceptable radiation levels are achieved. Decontamination wastes from the service machine are piped to the hot waste system.

5.11 Top Nozzle Shield Plugs and Closures

Shield plugs are installed in the reactor vessel top nozzle to maintain the integrity of the top biological shield to permit access to the service machine room. Each shield plug, fitted with a breech type closure, can be installed by the service machine. The plugs are designed to reduce radiation streaming up the reactor nozzles and the capture gammas formed within the shield to a dose rate of 0.75 mrem/hr measured at the level of the service machine room floor.

The shield plugs which will be in place during reactor operation include ferrophosphorus concrete for neutron shielding. This shielding reduces the fast neutron flux to $1_n/\text{cm}^2 \cdot \text{sec}$; the equivalent dose rate is less than

0.15 mrem/hr. This shield also assures adequate reduction of gamma radiation originating in the core, reactor pressure vessel, and the biological shield. The minimum thickness of concrete is 5-1/2 ft and occurs in the control rod plug where all available space above the concrete is required for the rod drive mechanism and a five inch thick lead shield. The lead shield is necessary since the activated control rod cable is stored in a drum located above the primary concrete shielding within the plug. All other plugs have a minimum of 9 ft of concrete.

Penetrations through the concrete shield for installation of thermocouple leads, control rod cables, locking mechanisms, drives, etc., are spiraled or offset to reduce the neutron streaming to acceptable levels. Annular streaming is reduced by designing for a maximum gap width of 1/8 in. and by providing two offsets in the annular path. At least three feet of concrete extend above the first offset.

The breech closure which is located at the upper end of all reactor shield plugs is shown on Figure 5.11. The closure consists of three major pieces: the outer spline ring, the ball screw nut, and the nozzle closure plug. The outer spline ring is free to rotate about the ball screw nut on two rows of ball bearings. The ball screw nut is in turn coupled to the nozzle closure plug by the ball screw. The entire assembly is locked and sealed to the reactor top nozzles by rotating the outer spline ring. The spline segments of the ring move under the matching spline segments on the nozzle extension.

The gas seal is effected by turning the ball screw nut which moves the nozzle closure plug upward. This upward movement compresses a gasket located between the plug and the outer spline ring. Electrical connectors pass through the center of the closure on the control rod drive shield plugs and the instrumented fuel column shield plugs.

6. AUXILIARY SYSTEMS AND PERFORMANCE

6.1 Vessel Cooling System

6.1.1 System Performance

The vessel cooling system shown in Figure 6.1.1 can simultaneously supply helium for cooling the following:

- a. Reactor vessel nozzle annuli and temperature barrier annulus - 12,000 lb/hr
- b. Reactor vessel nozzle interiors - 5700 lb/hr
- c. Experimenters' through tube bottom nozzles - 700 lb/hr
- d. The charge machine - 4200 lb/hr
- e. The service machine - 1250 lb/hr.

Most of the helium for the above usages is extracted from the reactor coolant system inlet piping. In addition, small flows enter the vessel cooling system from the blower seal leak-off and the reactor coolant recovery and purification systems (800 lb/hr), and from the burst slug detection system (210 lb/hr). Two centrifugal compressors, one operating and one stand-by, circulate the coolant gas. Both vessel cooling compressors are connected to the emergency 480 v load center. Failure of the operating compressor to maintain a pre-set pressure rise automatically starts the stand-by compressor.

Helium coolant is extracted from each of the two reactor coolant system loops, upstream from the reactor inlet coolant isolation valves. The flow in each line to the suction of the vessel cooling compressors is indicated on local panels and can be balanced by valve operation. These valves, which have position indicators on their corresponding main control room panel, can be closed or opened from the main control room by the reactor operator. An abnormally high flow in either suction line to the operating vessel cooling compressor is detected by a differential pressure switch across the respective flow element (FE-16-19 and FE-16-20). On high flow, each of these switches initiates closure of the respective control valve by a four-way solenoid valve.

The flow control valves in the supply lines from the compressors modulate to match the operating compressor characteristics. As one or more supplies are shut off, the flow control valves in the other systems partially close to reduce the total coolant flow.

The control valves in the helium supply lines to the reactor vessel top nozzle annuli, nozzle interiors, charge machine, and experimenters' through tube bottom nozzles have an additional function. A high differential pressure across any one of these three control valves closes that valve. Thus, any significant line rupture downstream of the control valve is isolated to prevent loss of reactor coolant. In the lines supplying helium to the service machine, this isolating function is accomplished by excess flow valves. In the line supplying the charge machine, a flow control valve and a pressure control valve located in series, receive signals to close from downstream pressure switches.

a. Nozzle Annuli

Flow to the reactor vessel temperature barrier annulus enters through the reactor vessel top nozzle annuli (Section 4.5.9). The temperature barrier coolant flow is distributed to the annulus of each reactor top nozzle by lines from secondary headers. The secondary headers are supplied by lines from a main header.

The lines from the secondary headers supplying helium coolant to each nozzle have check valves which prevent loss of coolant should the line rupture upstream of the check valve. Each individual nozzle coolant header has a flow element and globe valve to permit flow balancing.

A common flow alarm is annunciated in the main control room by abnormal flow in any of the secondary headers. The main header has a flow control valve and flow sensor. An alarm and flow indicating controller, associated with the main header, are located in the main control room.

The temperature of the reactor vessel top head is monitored by 46 thermocouples located between the top nozzles on the outside of the vessel. The temperature at each of these points is recorded by a logging system approximately once every ten minutes.

b. Nozzle Interiors and Experimenters' Bottom Nozzles

The flow to the interior of the reactor vessel top head nozzles and to the experimenters' through tube bottom nozzles is controlled in a manner similar to that described above for the nozzle annuli flow.

c. Charge and Service Machine Supplies

The coolant flow system for the charge machine and the service machine are similar. The flow passes through a flow element and a flow control valve. A flow indicating controller, located in the charge and service machine control room, permits manual adjustment of the control valve setting. A flow alarm annunciates if there is abnormal flow. In the charge machine system, operation of the flow control valve is interlocked with the position of the transfer tube; flow cannot start until the transfer tube is in the proper position and flow stops automatically when the transfer tube is withdrawn.

d. Nozzle and Charge and Service Machine Coolers

Coolant gas to the reactor vessel top nozzle interiors, the service machine and the experimenters' through tube bottom nozzles is cooled to approximately 125 F. Standby cooling capacity is provided. The temperature of the helium leaving the cooler is indicated in the main control room. An alarm annunciates in the main control room on high exit coolant temperature.

The piping upstream and downstream from the coolers contains three-way valves which are interlocked so that it is not possible to simultaneously block both discharge ports. These valves automatically divert gas from the operating cooler to the stand-by cooler upon loss of cooling water supply to the operating cooler.

Coolant gas supplied to the charge machine is cooled to approximately 350 F.

6.1.2 Component Design

a. Vessel Cooling Compressors

Each of the two vessel cooling compressors can deliver 23,000 lb/hr of helium at 315 psia. The casings are fabricated from steel plate. The shafts have static seals which are engaged when the running seals are not in operation. The shaft seals are similar to those described in Section 6.13. Each compressor has alarms which signal under-speed, high bearing temperature and high oil temperature.

b. Nozzle and Service Machine Coolers

There are two cooler units (a single cooler and an equivalent pair of smaller coolers) one operating and one on stand-by. The helium side of the coolers is of all-welded construction in conformance with the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII, including paragraph UW-2(1), and conform to the requirements of the standard helium leak test(2). Both the helium and water sides are protected by relief valves. The helium side of each cooler can be vented to the gas vent system and can also be evacuated by the vacuum system.

c. Charge Machine Cooler

A cooler is used to cool the gas supplied to the charge machine from 510 F to 350 F. This cooler is similar to the coolers used in the supply lines to the nozzle interiors and the service machine.

6.1.3 Emergency Operation

a. Stand-by Compressor Operation

Failure of the operating compressor to maintain a minimum pressure differential automatically starts the stand-by unit by a differential pressure signal. Shaft seal water and cooling water run continuously to both the stand-by and operating units. Isolation valves on each side of the stand-by unit are normally open. A check valve in the discharge of each compressor prevents back-flow through the stand-by unit.

b. Failure of Normal Electrical Power

Both the operating and stand-by vessel cooling compressors are connected to the emergency load center. In event of a loss of normal electrical power one compressor, which is previously selected by the reactor operator, will start automatically when power is restored to the emergency line by one diesel generator. Once the compressor has started, a circuit is established which allows automatic closing of each reactor coolant circuit inlet valve.

A separate cooling supply to the service machine is also activated in the event of low coolant flow to the service machine when it is in operation. A supply line from the yard high pressure helium storage tanks supplies helium to the service machine through a valve actuated by a flow signal in the coolant supply line to the service machine. A minimum flow of 1000 lb/hr, for a minimum of one hour, is assured by the emergency supply. When the vessel cooling compressor is restarted, the valve will close since the compressor will re-establish normal flow. The valve controlling the supply of helium from the yard storage tanks will close automatically on low reactor coolant system pressure. This will prevent containment shell overpressurization due to the injection

taken to regenerate this dryer tower. While one tower is removing water, the other tower can be regenerated. Regeneration is accomplished by diverting approximately 250 lb/hr of helium at 500 F and 300 psia from the reactor coolant system in a counter-current direction through the tower to be regenerated. An electric timer closes valve ECV-15-62 after a pre-set time. The hot, moisture-laden regeneration gas leaving the dryer flows to the blower seal leak-off recovery system upstream of coolers C-20, then to separator G-11-1 where the bulk of the water is removed. The hot regenerated tower is then cooled by diverting approximately 130 lb/hr of effluent helium from the active drying tower through the regenerated tower in a counter-current direction. Both hot and cold counter-current regeneration flow quantities are measured by flow element FE-15-61.

b. Carbon Dioxide Adsorber

The vessel G-6 contains 400 lb of Linde Type 5A molecular sieve pellets, to adsorb carbon dioxide. When the adsorbent bed becomes saturated with carbon dioxide, as indicated by instrument ACO₂E-13-17, the unit is valved off and regenerated. Regeneration is accomplished by directing plant air, heated by an external electric heater, through the adsorber and into the gas vent system. When the electric heater is turned off, the plant air continues to flow through the adsorber to cool the pellet bed. After the cooling of the bed has been completed, the remaining air is removed by the vacuum system. The adsorber is then ready for further carbon dioxide removal.

c. Helium Filters

The filters are of the porous metal type having replaceable filter elements. Each filter can handle the entire purification system flow. These filters are capable of retaining 95% of all particles whose two largest dimensions exceed two microns.

d. Carbon Monoxide and Hydrogen Converter

Carbon monoxide, hydrogen, and some hydrocarbons are catalytically combined with oxygen in this unit to form water and carbon dioxide. Less than stoichiometric quantities of oxygen are metered into the incoming gas. The oxygen system, protected by a pressure relief valve, is designed to prevent accidental release of large quantities of oxygen into the helium system. The oxygen is supplied from a high pressure cylinder through a pressure reducing valve. The oxygen piping system is cleaned of all organic matter in accordance with standard oxygen handling procedures. The flow of helium and oxygen are measured by flow elements FE-15-14 and FE-15-3 respectively. Temperature indicator, TI-15-2, with its associated temperature alarm, TA-15-2, is on the downstream side of the converter. The temperature of the reaction can be controlled by regulating either the quantity of oxygen added or the flow of cooling water through the water jacket of the converter.

The composition of the converter effluent gas mixture is determined by gas analyzers, ACO₂E-13-17, ACOE-13-18, and AH₂E-13-13. The effluent gas is checked for oxygen content by laboratory sample analysis.

e. Low Pressure Leak-Off Pre-Cooler

This heat exchanger unit C-21 is all-welded on the helium side and is designed and constructed in accordance with the ASME Boiler and Pressure Vessel Code.⁽¹⁾

f. Low Pressure Leak-Off Compressors and Knock-Out Drum

These units are similar in design and construction to the equipment described under Section 6.4.4.

6.2.4 Shielding of Components

→ The reactor coolant purification system equipment is located on the north side of basement floor levels 1 and 2 of the reactor building. The radioactivity level of the purification system depends upon the radiation level of the reactor coolant gas. Each component is monitored periodically with a portable radiation detector. Final shielding requirements will be determined on the basis of reactor operating experience.

6.3 Helium Vent System

6.3.1 Normal Gas Discharge System (Figure 6.3.1.1)

Piping and filters are installed to handle gas which is vented from vessels containing helium, from the vacuum pump discharge, and from small miscellaneous pressure relief valves. Flow indicators and globe valves permit variations in the venting rate consistent with atmospheric conditions and the radioactive contamination of the gas.

The largest flow to this system results from the manual venting of the reactor coolant system. The reactor coolant system blowdown valves are locked and require a key for operation. This blowdown flow is limited to 1350 lb/hr by orifice FX-14-24, located downstream from the controlled vent filters. The reactor coolant system can be completely vented in approximately three hours. Manual venting of the helium contained in the charge and service machines can be accomplished in less than one hour. Other manual vent flows are from miscellaneous vessels and the purging containers used for sampling the purification system gas. The maximum helium flow from these two sources is 150 lb/hr.

A means for depressurizing the reactor coolant system by remote manual actuation of valves HCV-14-56A and HCV-14-57 from the main control room is provided.

→ The vented gas flows through a filter system and then through three-way valves ECV-14-32 and ECV-14-42 to the stack. These three-way valves are electrically interlocked with the controlled vent valves in the charge machine vent headers. The charge machine valves cannot be opened when these three-way valves are in the "divert" position and if the charge machine vent valves are open when the three-way valves are switched to the "divert" position, the charge machine vent valves automatically close. Venting of the service machine is by manually operated valves. In the event of reactor containment, manual venting will be
→ stopped immediately.

The vacuum pump J-16 (Figure 6.3.1.2) discharges approximately 300 cfm (1350 lb/hr) of air through the oil vapor remover G-24, the AEC-type filter G-19, the iodine remover G-18, and through the three-way valves ECV-14-32 and ECV-14-42.

Check valves are installed at numerous locations in the normal gas discharge system to prevent back-flow from one part of the system to another.

The stack effluent is continuously monitored for particulate material, gaseous and iodine radioactivity. If the radioactivity reaches a pre-set level an alarm sounds in the main control room. To assure continuous monitoring, the stack monitoring system is duplicated. Both of the stack monitoring systems have malfunction alarms which are independent of the process alarms. If the

NOTES:

- FOR PIPING & INSTRUMENTATION SEE REF NO 2
- STACK EFFLUENT MONITORED FOR:
(a) PARTICULATE ACTIVITY
(b) TOTAL GASEOUS ACTIVITY
(c) IODINE ISOTOPE ACTIVITY
- BOILER CODE 1 REFERS TO ASME BOILER AND PRESSURE VESSEL CODE SECTION I, POWER BOILERS PIPING CODE REFERS TO AMERICAN STANDARD CODE FOR PRESSURE PIPING, ASA B31.1-1955
- FOR STACK MONITOR CIRCUITRY AND CONTAINMENT SHELL INTERLOCK SEE SPEC TA FIG TA 19.3 AND REF 15

NO.	DATE	DESCRIPTION	APPROV.	APPROV.
1	11/24/59	GENERAL REVISION	NR	NR
2	12/1/59	GENERAL REVISION	NR	NR
3	12/1/59	GENERAL REVISION	NR	NR
4	12/1/59	GENERAL REVISION	NR	NR
5	12/1/59	GENERAL REVISION	NR	NR
6	12/1/59	GENERAL REVISION	NR	NR
7	12/1/59	GENERAL REVISION	NR	NR
8	12/1/59	GENERAL REVISION	NR	NR
9	12/1/59	GENERAL REVISION	NR	NR
10	12/1/59	GENERAL REVISION	NR	NR
11	12/1/59	GENERAL REVISION	NR	NR
12	12/1/59	GENERAL REVISION	NR	NR

CERTIFIED FOR CONSTRUCTION

NO.	DATE	DESCRIPTION	APPROV.	APPROV.
1	11/24/59	GENERAL REVISION	NR	NR
2	12/1/59	GENERAL REVISION	NR	NR
3	12/1/59	GENERAL REVISION	NR	NR
4	12/1/59	GENERAL REVISION	NR	NR
5	12/1/59	GENERAL REVISION	NR	NR
6	12/1/59	GENERAL REVISION	NR	NR
7	12/1/59	GENERAL REVISION	NR	NR
8	12/1/59	GENERAL REVISION	NR	NR
9	12/1/59	GENERAL REVISION	NR	NR
10	12/1/59	GENERAL REVISION	NR	NR
11	12/1/59	GENERAL REVISION	NR	NR
12	12/1/59	GENERAL REVISION	NR	NR

EGCR GAS VENT SYSTEM PIPING & INSTRUMENT DIAGRAM

KAISER ENGINEERS
DIVISION OF KENNY J. KAISER COMPANY
OAKLAND, CALIFORNIA

U. S. ATOMIC ENERGY COMMISSION
OAK RIDGE OPERATIONS OFFICE
OAK RIDGE, TENNESSEE

DWG NO. **5927-EGCR-P173**

NOTES
 1. FOR PIPING & INSTRUMENT LEGENDS - SEE REF #1
 2. VALVES ARE SHOWN OPEN FOR PRESSURIZING CHARGE MACHINE

NO.	REVISION	DATE	BY	CHKD
1	GENERAL REVISION	12/24/60	NR	NR
2	GENERAL REVISION	1/10/61	NR	NR
3	GENERAL REVISION	1/10/61	NR	NR
4	GENERAL REVISION	1/10/61	NR	NR
5	GENERAL REVISION	1/10/61	NR	NR
6	GENERAL REVISION	1/10/61	NR	NR
7	GENERAL REVISION	1/10/61	NR	NR
8	GENERAL REVISION	1/10/61	NR	NR
9	GENERAL REVISION	1/10/61	NR	NR
10	GENERAL REVISION	1/10/61	NR	NR
11	GENERAL REVISION	1/10/61	NR	NR
12	GENERAL REVISION	1/10/61	NR	NR

CERTIFIED FOR CONSTRUCTION

EGCR		EVALUATION & HELIUM TRANSFER SYSTEM	
PIPING & INSTRUMENT DIAGRAM		KAISER ENGINEERS	
DIVISION OF HENRY J. KAISER COMPANY		OAKLAND, CALIFORNIA	
U.S. ATOMIC ENERGY COMMISSION		OAK RIDGE OPERATIONS OFFICE	
OAK RIDGE, TENNESSEE		SCALE: N/A	
DWG NO. 5927-EGCR-P128		FIG 6312	

gaseous radioactivity of the stack effluent reaches a dangerous level, containment isolation is effected automatically. The two diversion valves ECV-14-32 and ECV-14-42 form a part of the containment isolation system. The gas flow is diverted from the stack through the shield cooling system filters and into the containment building.

6.3.2 Pressure Relief Vent System

The pressure relief vent system consists of piping and filters which handle the discharge from pressure relieving devices on the reactor coolant system.

Two sets of three main safety valves, of which one set is a stand-by for maintenance, testing and reseating operation, relieve excessive reactor coolant pressure in the reactor vessel and the steam generators. Each of the valves is sized to handle the maximum credible steaming rate resulting from steam generator tube ruptures. Three valves, constituting one set, are set at slightly different relief pressures to minimize the possibility of simultaneous discharge. The reseating operation of relief valves takes place automatically when the reactor coolant system is depressurized to a predetermined level. If the relief valve fails to reseat depressurization is prevented by auto-isolation of the valve and switching in the stand-by (emergency) safety valves.

Rupture discs are installed downstream of the pressure relief valves to minimize leakage of helium. Leak-off connections between the pressure relief valves and the rupture discs prevent a pressure build-up caused by relief valve seat leakage. This leak-off flows to the low-pressure leak-off recovery system where it is monitored in the collection header. Abnormal flow actuates an alarm in the main control room. Restricting orifice FX-14-36 (Figure 6.3.1.1) prevents over-pressurization of the low-pressure leak-off receiver in the event of main pressure relief valve discharge.

Although the pressure relief valve leak-off lines lead to a common low-pressure leak-off receiver, it is unlikely that rupture discs of other systems will be affected by the build-up of back pressure from the discharge of the main relief valves because:

- a. The leak-off lines are $\frac{1}{4}$ in., schedule-80 pipe having very limited flow capacities, and
- b. The relief valve on the low pressure leak-off receiver is set to discharge at approximately 50 psia, whereas the rupture discs burst at approximately 100 psia.

6.3.3 Filters

Three types of filtering devices are used in the helium vent system; porous metal filters, AEC-type high efficiency filters, and silver-plated copper mesh iodine removers. In the controlled vent circuits each operating filter has a stand-by filter unit. The piping to these filters contains interlocked three-way valves which preclude simultaneous blocking of both discharge ports. Filter housings and elements are removable.

a. Porous Metal Filters

Both the main pressure relief valve circuit and the controlled vent circuit contain porous metal filters. These filters have sintered stainless steel filter elements which combine high mechanical strength, thermal shock resistance, good filtering characteristics, and are fire-proof. These filters retain 95% or more of all particulate matter larger than two microns. The filter elements are replaceable and are contained in a pressure vessel designed and fabricated in accordance

with the ASME Boiler and Pressure Vessel Code, Section VIII, including paragraph UW.(1)

b. AEC-Type Filters

The controlled vent circuit and the vacuum pump discharge circuit contain AEC-type high efficiency filters. These filters retain 99.97% or more of all particles 0.3 micron in size and larger, as measured by the MIL-STD-282 dioctyl phthalate (DOP) test.(11) The filter elements are incombustible and are resistant to moderate wetting.

c. Iodine Removers

All gases leaving the vent system pass through iodine removers. These removers consist of packs of silver-plated copper mesh.

6.3.4 Shielding of Components

The helium vent system is located on the reactor building ground floor. The vent system is surrounded by a 1-ft thick concrete wall, which forms a partial shield for the corridor area. A two-inch thick lead shield is also installed around the filters. However, since high activity will not exist continuously, the filters can be replaced when necessary. Additional shielding requirements for this system will be determined from reactor operating experience. Space has been allocated to permit additional shielding to be added at a later date, if operating experience indicates necessity for added shielding.

6.4 Helium Transfer, Evacuation, and Storage Systems

6.4.1 Helium Transfer System

a. General

The helium transfer system is designed to:

1. Pressurize the charge machine and service machine after their connection to the reactor vessel and to recover most of the helium contained in these machines after they are isolated from the reactor vessel
2. Transfer helium from the reactor coolant system to storage inside the reactor containment building
3. Unload helium from and return helium to the main helium storage area outside of the containment building.

b. Charge or Service Machine Pressurization (Figure 6.3.1.2)

→ The charge and service machines can be pressurized individually or simultaneously, but cannot be depressurized simultaneously nor can one machine be pressurized while the other is being depressurized.

→ After either or both of the machines have been evacuated, helium flows from the manifolded high pressure storage cylinders F-4 and very high pressure storage cylinders F-59 through pressure reducing valve PCV-17-5 to the respective machines. The block valves HCV-17-8 and HCV-17-98 on the high pressure and very high pressure cylinder manifolds and the high pressure line block valve HCV-17-6 can be operated from, and their position noted on, panel K-6 located in the charge and service machine control room. The setting of pressure reducing valve PCV-17-5 can be remotely adjusted by the indicating controller PIC-17-5 on panel K-6.

c. Transfer of Helium from the Reactor Coolant System to Storage

After the charge machine has been connected to the reactor vessel and the vessel cooling flow into the charge machine has started, the pressure in the reactor coolant system may increase. Pressure switch PS-16-156A (Figure 6.1.1) senses reactor coolant system pressure increase and at an adjustable pre-set value, automatically opens valve PCV-16-156 (Figure 6.3.1.2). This permits any excess gas in the reactor coolant system to flow into the manifolded low pressure helium storage tanks F-10. These tanks are initially at a pressure of 20 psia. When helium starts flowing into tanks F-10, the operator in the charge and service machine control room remotely opens block valve HCV-17-7 and starts the transfer compressors J-12 to divert flow past the low pressure helium storage tanks to the transfer compressor and helium coolers. The transfer compressors discharge to the high pressure helium storage cylinders F-4 and the very high pressure helium storage cylinders F-59. When the reactor coolant system pressure returns to normal, PCV-16-156 automatically closes. The position of the valve PCV-16-156 is indicated on the main control room panel and on panel K-6. This valve can also be remotely operated from the main control room by hand switch HS-16-156. The pressure in the low pressure helium storage tanks is indicated both locally and on panel K-6. High pressure in the tanks actuates an alarm. The block valve on the line to outside storage is normally closed and requires a key for operation to prevent the inadvertent flow of contaminated helium to outside storage. The key is kept in the charge and service machine control room.

d. Depressurization of Charge or Service Machine

After the charge or service machine has completed its operations, a sample of the contained gas is analyzed for radioactivity. This sample flows through valve HCV-17-47 or valve HCV-17-49, which are operable from the charge and service machine control room. Activity is measured by radioactive monitor RE-20-11-9 which is connected to indicating controller RIC-20-11A and to alarm RA-20-11B in the charge and service machine control room. If the activity of the sample is above a predetermined value, the sample is vented to the stack through three-way valve RCV-20-11. The gas in the machine is then diverted through the gas vent system if atmospheric conditions are favorable.

If the initial activity of the gas in the machine is within acceptable limits, the gas sample is diverted to low pressure storage tanks F-10. Pressure is then equalized between the machine and tanks F-10, with the gas flowing past radioactivity monitor RE-20-11-10. After pressure equalization, tanks F-10 are valved off and the transfer compressors remove the remaining gas from the machine. A pressure switch on the compressor suction automatically stops the compressor when the pressure is reduced to approximately 20 psia. This pressure limit is set to preclude air in-leakage which might occur if the pressure were reduced to a value below atmospheric.

If, during depressurization, the activity of the flowing gas rises above a pre-set level, the control circuit automatically closes the block valves and stops the compressor, if it is running.

6.4.2 Gas Evacuation System

The evacuation system, consisting of a vacuum pump, valves, piping, and an oil vapor remover, removes air or helium from vessels and piping of the reactor coolant system and auxiliary systems (Figure 6.3.1.2). To protect the operator,

certain piping connections at the charge and service machine which may contain contaminated helium are evacuated after being valved off and prior to disconnection from a charge or service machine nozzle.

6.4.3 Helium Storage

A three-tier rack of eighteen cylinders (Figure 6.4.3) is located approximately 150 ft north of the reactor building. These cylinders F-24 are used to store clean helium for:

- a. Reactor coolant system fill and make-up
- b. Charge and service machine helium make-up
- c. Burst slug detection system purge.

→ Helium make-up to the reactor coolant system flows through a pressure reducing valve which is protected by a relief valve, through a flow element, a check valve, and then through a block valve manually operated from the main control room. Helium make-up to the reactor coolant system enters upstream from the dryer in the gas purification and recovery system. The pressure of the helium supply upstream from the pressure reducing valve is indicated in the main control room. Make-up flow is indicated and totaled by a flow integrator in the main control room. The make-up piping system is designed to limit flow to a maximum of 120 lb/hr, which precludes sudden over-pressurization of the reactor coolant system.

→ Very high, high and low pressure helium storage tanks are installed within the containment building. Gas from the charge machine, service machine or reactor coolant system is stored temporarily in the low pressure tanks F-10 until the
→ transfer compressors can transfer the gas to the very high and high pressure cylinders. The large volume of the low pressure storage tanks facilitates rapid depressurization of the charge and service machines.

→ The high and very high pressure cylinders contain ample helium to pressurize simultaneously both the charge machine and the service machine. Helium make-up from outside storage can be routed directly into the high pressure or very high pressure cylinders or can be routed into either system through the transfer compressor.

A separate helium storage facility, F-58, is provided exclusively for emergency cooling of the service machine and the emergency cooling loop reactor nozzle. It consists of a Bureau of Mines helium tank car permanently mounted adjacent to the outside helium storage cylinders, F-24. Helium for emergency cooling of the service machine flows at 1000 lb/hr from F-58 through a pressure reducing valve and a flow control valve directly to the service machine and at 200 lb/hr to the emergency cooling loop reactor nozzle.

6.4.4 Components

→ All pressure vessels, other than the very high and high pressure cylinders, in the evacuation, helium transfer and storage system, are designed and constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, including paragraph UW-2.⁽¹⁾ The very high pressure helium storage cylinders, F-58, are constructed in conformance with ICC-107A-3750 specifications and storage cylinders, F-59, with ICC-3A-2400. The high pressure storage cylinders F-4 and F-24 are designed and constructed in conformance with Military Specification MIL-C-7127(Aer).⁽¹⁴⁾ All vessels and equipment are protected by relief valves. Those inside the containment building discharge into a closed piping system which vents to the gas vent system. All vessels can be evacuated by the vacuum system and conform to the requirements of the standard helium leak test⁽²⁾.

HELIUM TANK CAR MHA# 1012 30 CYLINDERS
39000 SCF TOTAL (70°F)
2115 PSIA WORKING PRESSURE
1478 ACTUAL CF TOTAL 20-~~0000~~ MIN. PRESS
(F 58)

12 000 SCF TOTAL (70°F) 8.5 PSIA OPERATING PRESSURE
20 000 ACTUAL SCF TOTAL 20.5 PSIA MAXIMUM PRESSURE

- 2 CYLINDERS F-24 CONN ARE 1" O.D X 11 GAUG
(120) WALL STEEL TUBING, SEAMLESS-
ASTM A 192 30 T COLD DRAWN
AND ANNEALED "DEAD SOFT"

1. *Explain the importance of the following factors in the development of a country's economy:*

EGCR
HELIUM STORAGE FLOW
PIPING & INSTRUMENT
DIAGRAM
KAISER ENGINEERS
 DIVISION OF HEWITT & KIRK COMPANY
 OAKLAND, CALIFORNIA
FOR
U. S. ATOMIC ENERGY COMMISSION
 OAK RIDGE OPERATIONS OFFICE
 OAK RIDGE, TENNESSEE
 DRAWING NO. **5927-EGCR-C114**

a. Transfer Compressors J-12

Three diaphragm type, two stage compressors are installed in parallel. Normally, all three compressors operate simultaneously; however, the piping and valving system permits operation of only one or two of the compressors if required. This proven type of compressor uses intermittent oil pressure to actuate a flexible metal diaphragm which compresses the helium in a chamber on the other side of the diaphragm. No lubricant or other contaminant comes in contact with the helium being compressed. An elapsed time meter in each motor circuit indicates the total pumping time since the last diaphragm change and diaphragms are changed after every thousand pumping hours.

Hydrocarbon detector AE-17-90 (Figure 6.3.1.2) detects any diaphragm damage which results in oil entering the helium stream. In addition, a knock-out drum downstream from the compressors captures large quantities of oil in the event of diaphragm rupture.

b. Vacuum Pump J-16

The vacuum pump is a standard mechanical rotary type having oil seals. It has a displacement of approximately 300 cfm and an ultimate blank-off pressure of ten microns. A relief valve and strainer on the suction side protect the pump.

c. Oil Vapor Remover G-24

The oil vapor remover has a replaceable cartridge containing oil-adsorbent material.

d. Cooler C-17

The helium side of the cooler is of all-welded construction to minimize leakage. Flow of service water through the tubes is continuous.

e. Storage Tanks F-10

The storage tanks are of all-welded construction and conform to the ASME Boiler and Pressure Vessel Code. (1)

f. Storage Tanks F-4 and F-24

The high pressure storage tanks are fabricated by forging and machining and conform to Military Specification MIL-C-7127(Aer). (14)

g. Storage Tanks F-58

The very high pressure storage tanks in the yard are a permanently installed Bureau of Mines Helium Tank Car. The storage tanks are fabricated and tested to conform with ICC-107A-3750 specifications.

h. Storage Tanks F-59

The very high pressure storage tanks located inside the containment shell are from a compressed gas trailer. The storage tanks are fabricated and tested in accordance with ICC-3A-2400.

6.5 Shield Cooling System

6.5.1 Cooling Requirements

Essentially all of the nuclear radiation which leaves the pressure vessel is absorbed and converted to heat in the 10-ft thick concrete biological shield which surrounds the reactor. Most of this internally generated heat escapes from the concrete by conduction to the inside surface of the shield. From there it is removed by the shield cooling air stream. A small portion of the heat generated in the biological shield is dissipated from the outside of the shield to equipment rooms and is removed as part of the normal cooling load of these areas. The temperature of the shield is kept below 200 F.

6.5.2 System Design

The design is predicated on a maximum air supply temperature of 100 F. A total flow of approximately 42,000 cfm is circulated through the biological shield cavity which surround the reactor pressure vessel (Figure 6.5.2). Of this total flow, approximately 17,000 cfm is outside air which is filtered and distributed by an inlet duct directly into the shield cavity. The remainder, approximately 25,000 cfm, is drawn into the shield cavity from adjacent rooms. The shield cooling air is initially confined to a three-inch passage formed by a metal baffle mounted on the inner face of the shield and is then passed into the pressure vessel chamber. The air is drawn into an exhaust duct system and through filters by a blower and discharged through the stack to the atmosphere.

A bypass duct from the exhaust system to the supply, together with a temperature controller in the supply duct, permits temperature regulation of the

Pages 6-23 thru 6-50 are missing from document

compressor. Each sub-system includes a low pressure tank, a pump, two coolers and a high pressure tank.

In normal operation the flow from each pump is split into two streams. Part of the flow goes to the smaller cooler and the remainder bypasses the coolers. The flow through the bypass is controlled to maintain the temperature of the combined flows at 140 F. This combined flow passes to the high pressure tanks, to the filters and thence to the seals. Each pump has sufficient capacity to supply the requirements of all of the seals. Hence, approximately one-half of the pump discharge is recirculated to the low pressure tank to maintain a constant level in the high pressure tank.

If the influent temperature to the high pressure tank exceeds 140 F, the temperature control valve in the bypass line closes. If the small coolers C-49-1 and C-49-2 do not maintain the respective influent temperatures to the high pressure tanks at or below 140 F with the bypass line closed, the tandem three-way temperature control valves TCV-38-76B and TCV-38-77B divert the seal water flow through the large coolers C-39-1 and C-39-2. The tandem three-way valves also divert the cooling service water from the small to the large coolers at the time of the seal water diversion.

Water level in the low pressure tank is maintained by adding make-up from the condensate system. A low pressure helium blanket exists in the low pressure tank. This helium is supplied from the low pressure leak-off system.

The high pressure tank is pressurized with helium from the discharge of each of the two reactor coolant blowers. Dryers prevent water vapor from entering the reactor coolant stream.

Bypass demineralizers remove ionized corrosion and radioactive fission products. In addition, the pH is controlled by manual adjustment of the flows through the two demineralizers. The composition of the resin is different in each tower as required for pH control. Conductivity, radioactivity, and pH of the main seal water stream is continually monitored.

During normal operation the two seal water sub-systems operate independently; both pumps deliver full flow, and both half-capacity coolers are in service.

If a rupture occurs in the header from the high pressure tanks to the seals, the seals supplied by that system are lost and the affected compressors must be shut down.

A rupture of one of the high pressure helium lines to the high pressure tanks would be detected by high differential pressure across one or more orifices. High differential pressure in either direction actuates an alarm and the valve associated with an orifice having high differential pressure closes.

If one pump fails, or a line from a pump to a high pressure tank ruptures, the loss of flow to the tank actuates an alarm and a bypass valve opens, thus supplying both sets of seals from one sub-system.

High or low water level in the high pressure tanks actuates an alarm and high level shuts down the associated pump.

Failure of one pump results in an accumulation of water in the associated low pressure tank. A level control valve opens on high water level in either low pressure tank, to supply both systems through one pump. A failure in a return line from a set of seals results in a low level in one of the low pressure tanks. A low water level switch shuts down the associated pump and actuates an alarm.

The low pressure tanks are vented to the low pressure leak-off recovery system to prevent air contamination of the seal water and to prevent the possible build-up of helium pressure in the low pressure tanks.

→ Upon loss of normal power, both pumps stop and the seals are supplied for a minimum of 30 min by the water stored in the high pressure tanks by opening of bypass valves in the high-pressure tanks' supply lines. The water is returned to the low pressure head tanks until the tanks overflow. The normal level maintained in the low pressure tanks is sufficient for approximately five minutes operation of the seals, without pumps, before overflow to the warm waste system occurs. Prior to overflow, the level control valve opens to interconnect the two low pressure tanks. One of the two pumps starts when emergency power becomes available and all seals are supplied by this pump.

The filters are arranged in four sets of two filters each. The filter piping is arranged with tandem three-way valves to prevent flow stoppage by operator error. High differential pressure across any filter actuates an alarm.

The dryers in the high pressure helium supply lines are self-regenerating. A rise in water level in a high pressure tank forces moist gas to the dryer, but a reduction in water level reverses the gas flow and regenerates the dryer. Because gas is absorbed by the water in the high pressure tank, the net flow of gas is toward the tank. Two small dryers, readily replaceable, are used in parallel in each helium supply header.

A supply header is installed to each reactor coolant blower (2 seals each), each vessel cooling compressor (1 seal each), and each blower seal leak-off compressor (2 seals each).

The seals on the reactor coolant blowers and on the vessel cooling compressor are identical.

All water supplied to the seal is controlled by a differential control valve. This control valve maintains constant differential pressure across the inner bushing. A flow element generates an overriding signal to limit the flow through the valve to 200% of normal design flow. The flow downstream from the control valve divides, with the major stream entering the seal and the remainder entering the cooling gland. The flow to the cooling gland is manually controlled.

A mixture of water and helium leaves the seal at the inner drain. This mixture is separated from the helium flow to the blower seal leak-off coolers and is limited to 3 cfm by a flow control valve. A level control valve controls the rate of water flow return to the seal water supply system.

Part of the flow to the seal inlet chamber is bled off to control the temperature in the bushing.

→ The outer primary drain is maintained at a pressure of 5 to 10 psig. The atmospheric drain is normally discharged to warm waste.

Alarms are included for high seal water supply flow, high inner drain water flow, high helium flow, high water level in the separating section, and high water flow from the outer primary drain.

A pressure switch closes a set of contacts in the blower drive motor starting circuits to prevent starting of the blower without seal water pressure at the seal. The blower seal water permissive in the main blower starting circuit can be manually bypassed. This manual bypass may be required if the reactor system has been depressurized.

The seals for blower seal leak-off compressor J-14-1 and J-14-2 are similar to those for the reactor coolant blowers and vessel cooling compressors. This arrangement is typical for both blower seal leak-off compressors. The flow of seal water to the seals on both ends of the shaft is controlled by a differential pressure control valve. A flow element transmits an overriding signal to limit the flow through the differential pressure control valve to 200% of design flow. The primary outer bushing drain is maintained at a pressure above atmospheric by connection to the low pressure head tank. The atmospheric drain is connected to warm waste. The flow from the casing drain is controlled by a level control valve. A pressure switch in the seal water supply line closes a set of contacts in the compressor drive motor running circuit to prevent operating the compressor without seal water at the seals.

6.13.2 System Components

a. Low Pressure Tanks

The low pressure tanks F-52-1 and F-52-2 are designed for an internal pressure of 15 psig. The volume of tank F-52-1 is approximately 2300 gal. The volume of tank F-52-2 is approximately 3800 gal. The tanks are helium tight and bear ASME code stamps.

b. High Pressure Tanks

The high pressure tanks F-53-1 and F-53-2 are designed for an internal pressure of 350 psig. The volume of each tank is approximately 4500 gal. The tanks are helium tight and bear ASME code stamps.

c. Pumps

The seal water pumps J-54-1 and J-54-2 are each designed to deliver a maximum of 225 gpm at a discharge pressure of 410 psig.

d. Coolers

The coolers are designed for 530 psig pressure on the shell (seal water) side and 70 psig on the tube side. Coolers C-39-1 and C-39-2 are each designed to cool 120 gpm from 180 to 100 F. Coolers C-49-1 and C-49-2 are each designed to cool 80 gpm from 160 to 100 F. All shells have ASME code stamps.

e. Filters

The filter casings are designed for 400 psig pressure and 65 gpm flow. The filters remove 98% of all particulate matter 5 micron or larger. The piping from the filters to the seals and all seal parts are fabricated from corrosion resistant materials.

f. Dryers

The dryer casings are designed for 350 psig. Each dryer has a water removal capacity of five pounds prior to regeneration.

g. Demineralizers

The demineralizer design pressure is 100 psig. Each demineralizer contains a minimum volume of 3 ft³ of resin. One column produces an effluent having a pH of 7.0 ± 0.2 and the other column produces an effluent of pH 11.0 ± 0.4. The unit has a capacity of 12,000 grains of equivalent calcium carbonate per cubic foot of resin. The design flow is 10

gpm per column.

6.14 Containment Shell Spray Cooling System

6.14.1 General Requirements

a. Emergency Operation

During an accident situation in which the containment shell is isolated, the heating and ventilating system, which normally removes heat from the building, is valved off. The external surface of the containment shell is sprayed with water, during such an accident, to remove heat and prevent the pressure and temperature inside the shell from exceeding the design conditions. The spray system also reduces the pressure inside the shell after an accident and effects a corresponding reduction in the activity leakage to the environment.

b. Summer Cooling

It may be necessary to operate the spray cooling system for certain periods during the summer to remove solar heat absorbed by the shell. The capacity of the reactor building heating and ventilating system, which is based on an insulated containment shell, probably will be inadequate to avoid temperatures above 105 F inside the reactor building unless some exterior spray cooling is utilized.

6.14.2 System Design

The design is based on a heat removal of 32 million Btu/hr and a total spray water flow of approximately 2500 gpm at the normal maximum river water temperature of 71 F.

The water is supplied from the fire protection water system through a valve house in which strainers, valves and control devices are located. The system downstream of the valve house consists of three 6-in. headers, each of which is branched to a number of ring headers with spray nozzles (Figures 6.14.2.1 and 6.14.2.2).

Most of the water is distributed over the dome using solid-cone spray nozzles, and the remainder of the water is sprayed on the vertical portion of the shell, concentrated near the top, using flat spray nozzles.

The sprayed water flows down the dome and the sections of the vertical shell that have no interior concrete wall. The water flow is directed over each stiffener ring and back against the shell by means of baffles. The water drains from the sides of the shell over the roof of the loop cells and into the storm sewer system.

Automatic flushing of strainers during the full operation of the system is provided since the accessibility by personnel for manual flushing, following the MCA or other serious depressurization accidents, is questionable and clogged strainer baskets could seriously affect the capability of the system.

6.14.3 System Performance

The piping and instrumentation for the containment shell spray cooling system is shown on Figure 6.11.6. A containment isolation signal opens valves PdCV-21-38A and 38B and initiates water flow through both strainers to the containment shell spray headers and nozzles.

The maximum pressure condition reached inside the shell following the MCA is 8.9 psig. When this maximum pressure is reached, the steam-gas mixture within the shell is at an average temperature of 165 F, the highest temperature of the containment shell wall is 105 F, and the temperature rise of the sprayed water is approximately 29 F. The external spray system has the capability of removing 32 million Btu/hr from the shell with the temperature of the gases and steam inside the shell at 165 F even when the automatic flushing of strainers is in progress. A bypass with 4-in. globe valves is provided in the valve house. Spray water for summer cooling is provided by manually throttling one of these valves. A strainer has to be flushed manually during this operation. Emergency operation of the spray cooling system, including automatic strainer flushing, would proceed irrespective of whether the system was being used for summer cooling at the time the accident occurred.

6.15 Emergency Cooling System

6.15.1 General

The emergency cooling system can be used to cool the reactor core when the reactor coolant system is depressurized and the reactor core is isolated from the coolant loops. Core cooling must be available for decay heat removal for approximately 90 days after reactor shutdown. The emergency cooling system, in conjunction with the reactor coolant system and the vessel cooling system assure this continuity of decay heat removal. The availability of cooling equipment following certain accidents which could affect cooling equipment is described in Sections 8.10.6 and 8.11.7.

The emergency cooling system consists of the following independent sub-systems:

- a. Heat removal system
- b. Purge gas system
- c. Fission product removal system.

The heat removal system is used to transport heat from the reactor core to the heat removal system heat exchanger where the heat is transferred to cooling water.

The purge gas system is used to blanket the reactor core graphite with nitrogen to minimize the oxidation of the core graphite until core temperatures are reduced to a level at which an unlimited supply of air would not support self-sustained oxidation. It also will serve to dilute any hydrogen formed below flammable mixtures in any volume of air.

The fission product removal system is used to prevent over-pressurization of the containment shell due to the continuing nitrogen purge which ultimately discharges into the reactor building.

The principal components of the heat removal system are installed in an experimenter's cell just outside the reactor building. This cell serves as containment for this equipment and any failure within the loop equipment therefore could not result in violation of containment philosophy.

The general arrangement of the systems and flow paths are indicated in Figure 6.15.1.

6.15.2 Systems Design and Performance

The heat removal system is designed to circulate helium, steam, nitrogen, air, or any mixture thereof at pressures from 14.4 to 24 psia. The heat removal capability of the system is compatible with core cooling requirements starting

one hour after an accident to prevent runaway graphite oxidation and additional fuel failures. The pressure-containing parts of the system are designed for an internal pressure of 350 psig, however the system is not capable of operation at this pressure. This is due to the large power requirement which would be necessary for emergency cooling compressor operation of 350 psig. Inadvertent pressurization of the heat removal system to reactor coolant system pressure will not result in failures in the system. The system is isolated from the reactor coolant system until it is required for core cooling service, which can be effected only when the reactor is shut down and the reactor coolant system is depressurized. Therefore, the design is such that credible failures of the reactor coolant system cannot impair the heat removal capability of the system. Conversely a failure which originates within the heat removal system cannot result in failure of the reactor coolant system.

The heat removal system is arranged so that the operator can periodically test the compressors, heat exchanger, and other equipment within the cell by opening the bypass valve.

A test of the complete heat removal system can be performed; however, this would require reactor shutdown and depressurization, isolation of the two reactor coolant loops, and opening of the block valves in the lines to the equipment within the cell.

The purge gas system is designed to enable introduction of a nitrogen gas purge into the emergency cooling system at an initial rate of 1000 scfm for 1½ hours followed by a purge of 200 scfm for the next three days. Liquid nitrogen is bled into a vaporizer to assure that only nitrogen in gaseous form enters the emergency cooling loop. The liquid nitrogen storage tank is installed external to the reactor building and the experimental cell. The nitrogen vaporizer is installed in the utility tunnel external to the experimental cells.

The reactor building fission products removal system is installed in a separate experimental cell. The system is designed to permit the controlled venting to the stack of the reactor building environment following accident conditions at a rate at least equal to the long term nitrogen purge rate. There is adequate time available for the operator to take deliberate and considered action to initiate controlled venting from the containment shell without any danger of overpressurization due to the continued nitrogen purge. The system includes filters which limit the release of radioactivity and particulate matter from the reactor building as the result of an accident.

The design of the systems is such that the systems can be continuously operated without maintenance for a period of at least three months. To assure continued core cooling, and to permit such direct maintenance as can be performed under accident conditions, the two compressors and the heat exchanger in the heat removal system are shielded from each other by dry stacked concrete block walls.

The experimental loop cell which contains the major components of the heat removal system is a concrete structure with a 3/16 in. thick metal lining to assure leak tightness and containment of activity released within the cell. The cell is designed for an internal pressure of 12 psig at a temperature of 150 F. A 33-in. diameter rupture disk between the cell and the reactor building assures that the cell design pressure will not be exceeded.

The roof of the cell has removable hatches, however, access through these hatches requires cutting of the cell liner. Personnel access to the cell is gained through a portable airlock which is attached to the personnel hatchway in the cell roof.

The leak-tightness of the cell is being determined by pressure tests similar to

those used for the containment shell. Cell leakage rates at design pressure of 12 psig will be less than 0.5% of contained weight of air per day for the cell without penetrations and less than 1%/day for the cell after installation of the penetrations.

6.15.3 Components

a. Piping

A jacketed pipe system between the reactor vessel top nozzle and the reactor vessel bottom nozzle in loop position TL-1 and the experimental cell is used to connect the heat removal equipment to the reactor core. The use of the jacketed pipe system minimizes the effect of failures in the internal pipes since the piping jacket affords continuity in flow paths.

The piping system associated with the nitrogen purge supply penetrates the cell containment and terminates in connections to the piping and each compressor.

b. Compressors

Each of the two compressors have a capacity consistent with the flow requirements indicated on Figure 6.15.1. Each compressor is electric motor driven through a speed increasing gear. The electric motor capacity of 300 hp is adequate for the design pressure, flow, and coolant compositions. Power to the compressors is from the 480 v emergency power supply. The operation of the compressors is controlled by remote manual means from the main control room.

c. Heat Exchanger

The heat exchanger in the heat removal system is a shell and tube type heat exchanger with water in the tubes and coolant in the shell. The tubes are so arranged that shell expansion joints are not required. The heat exchanger serves to cool the coolant from the reactor core and also to condense moisture or water vapor in the coolant stream. Condensate traps assure that the heat exchanger gas side will not become flooded.

d. Nitrogen Supply

The nitrogen storage consists of a supply of liquified nitrogen. A circulating water vaporizer is used to assure that the liquid nitrogen is vaporized and that only gaseous nitrogen enters the heat removal system. The vaporizer can be controlled by remote manual controls located in the main control room.

e. Filters

The filters in the containment shell filtered vent system include the following:

1. Roughing filter

This filter is used to remove large particulates and is at least 95% efficient for the removal of particulates 5 micron size and over.

2. Charcoal trap

The charcoal trap is used to remove radioactive iodine from the vent and is at least 95% efficient.

3. Absolute filter

Two absolute filters are used in the system. These filters are

located before and after the charcoal trap and are used to remove remaining particulate from the vent stream. These filters are at least 99% efficient for the removal of particulates 0.3 micron size and over.

The filters are not designed to trap noble gases. The effect of these gases is considered in the radiological hazards calculations (Section 9).

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7. INSTRUMENTATION SYSTEMS

7.1 Reactor Safety System

7.1.1 General

The reactor safety system generates signals and takes automatic actions to avoid abnormal conditions which are unsafe or potentially unsafe. An unsafe condition is a condition which, if uncorrected, could damage plant equipment or result in radioactive exposure of operating personnel or the general public. Such conditions, without a safety system, could remain uncorrected because of absence of corrective action by the operator, an occurrence too fast for operator correction, operator error, or malfunctioning of automatic control systems. The safety system consists of the following:

a. Instrumentation and Control

Instrumentation and controls enable the operator to monitor and evaluate plant operating conditions.

b. Alarms

Alarms warn that a prescribed operating limit is being approached, and that corrective operator action should be taken or that the operating limit has been exceeded and, in some instances, that automatic safety action, such as a rod withdraw prohibit, has occurred.

c. Reactor Scram System

The scram system reduces the reactor power level when an operating limit has been exceeded either as a result of operator error, malfunction of plant automatic control systems, or failure of certain plant equipment and components.

d. System Interlocks

Interlocks are employed to prohibit certain control actions (either automatic or manual) which would result in an unsafe condition.

7.1.2 Criteria for Reactor Safety System Design

The following criteria were used in the design of the reactor safety system:

a. Independence

The reactor safety system and the plant control systems are completely independent. Each system includes its own detectors and circuitry; neither system is dependent upon the other for proper operation, nor does failure of any portion of one system affect the operation of the other system in any way.

b. Reliability

→ The basic reactor safety system components are relays and transistors. Relays are used for output circuit isolation, since a relay with multiple contacts can be used to control a number of isolated circuits.

→ Transistors are used for certain applications because of the following characteristics:

1. Ability to perform logic functions
2. Extremely fast response
3. Ability to supply variable gain
4. Good sensitivity
5. Small weight and space
6. Ability to supply variable frequency a-c amplification, if required
7. Ability to perform switching functions.

Transistors are used where sufficient speed of response cannot be obtained with other types of components, and where it is desirable to maintain consistency with transistorized circuits. Vacuum tubes are permitted where their application improves system performance, or when economic savings can be realized without sacrificing performance.

c. Testing and Accessibility

Most of the reactor safety system components are mounted in pull-out drawers located in the rear section of panels in the main control room. The equipment is accessible for testing and maintenance. Each channel is isolated so that failure of components in one channel does not affect any other channel. The safety system includes built-in test circuits to permit checking and testing of the system from, but not including, the detector to the actuator while the reactor is operating without interfering with the operation of the safety system.

d. Non-Interference with the Control Systems

The reactor safety system design, including selection of parameters and trip points, is such that the control system will control the plant during normal transients without interference from safety actions. Normal transients are defined by the temperature criteria for the fuel, fuel cladding, coolant piping, and vessels in the reactor coolant system.

7.1.3 Safety Actions

The reactor safety system continuously monitors operating conditions and automatically initiates safety actions upon deviation from preset values. No auxiliary power sources are required to assure the protection of scram action. Safety actions fall into the following four classes:

a. Alarms

Visual or audible alarms, or both, indicate approach to, or existence of, an unsafe plant condition.

b. Rod Withdraw Prohibit

This action prevents withdrawal of the reactor control rods, when a plant condition is reached which does not require a complete shutdown but does require that reactivity addition be delayed or stopped (Figure 7.1.3).

c. Manual Safety Insert

This action is operator controlled and is used by the operator to initiate a controlled shutdown; or to correct an abnormal condition; whenever possible, it should precede the more drastic manual scram. During a safety insert all rods are driven into the core at the speed of 2 in./min.

d. Scram

This action (either automatic or manual) is used when plant operating conditions require complete and immediate reactor shutdown. Power to the control rod drive clutches is interrupted, permitting the control rods to drop into the core by gravity. Manual reset of the control rod drive clutch power is required before rods can be withdrawn following a scram. Reset is only possible after the condition causing the scram has been corrected.

Table 7.1.3 indicates the trip points and safety actions in the reactor safety system.

TABLE 7.1.3

Safety System Scram Actions

<u>Parameter</u>	<u>Alarm</u>	<u>Rod Withdraw Prohibit</u>	<u>Scram</u>
Neutron flux (start-up range)	1 Mw	1 Mw	3 Mw
Flux minus flow	+5%	+5%	+10%
Reactor outlet temp.	1065 F	1065 F	1075 F
High coolant pressure	320 psig	--	325 psig
Low coolant pressure	--	225 psig	225 psig Intermediate & power range
Steam generator No 1 drum level	-7 in.	--	-9 in.
Steam generator No 2 drum level	-7 in.	--	-9 in.
Nozzle annuli flow	7500 lb/hr	--	6000 lb/hr
Nozzle interior flow	4000 lb/hr	--	3000 lb/hr
Loss of normal plant power	Yes	--	Yes
Nozzle interior coolant temperature	135 F	--	150 F
Loss of either blower	Yes	--	Yes

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In addition to the tabulated safety system actions, safety insert and scram functions can be manually initiated from the main control room; manual scram also can be initiated from the charge machine console and the service machine console.

7.1.4 Safety System Design

a. Nuclear Instrumentation System

The nuclear instrumentation system measures the reactor period and neutron flux which are required for safe operation of the facility. The system monitors the neutron flux from the source range through intermediate and power ranges. These signals are used for indication and safety, and for certain interlock actions, but are not used for automatic control purposes. The system consists of seven channels as shown in Table 7.1.4.1 and Figure 7.1.4.2.

TABLE 7.1.4.1

Nuclear Instrument Channels

<u>Channel Designation and Numbers</u>	<u>Number of Channels</u>	<u>Measurement</u>	<u>Channel Range</u>	<u>Neutron Flux Range</u>
Start-up (1 and 2)	2	count rate period	1 to 10^5 cps -1 to 10 dpm	1.4 to 1.4×10^5 $n/cm^2 \cdot sec$
Intermediate (3 and 4)	2	period neutron flux	-1 to 10 dpm < $10^{-4}\%$ to 150%	10^4 to 2.2×10^{10} $n/cm^2 \cdot sec$
Power (5, 6 and 7)	3	neutron flux	< 1% to 150%	10^8 to 2.2×10^{10} $n/cm^2 \cdot sec$

The detectors are positioned in instrument access tubes which pass through the biological shield as shown in Figure 7.1.4.1. Nine access tubes are installed around the reactor, but only seven are required for reactor operation. The two remaining tubes are spares.

The two instrument tubes containing start-up channels 1 and 2 detectors are equipped with graphite thermal columns which thermalize fast neutrons to permit proper measurement of neutron flux in the reactor during start-up. The start-up channel detectors are remotely positioned from the main control room so that the detectors can be withdrawn into the tubes to protect them while the reactor is operating in the intermediate and power ranges. This withdrawal also permits the start-up channel instruments to remain on-scale at all times. The intermediate and power channel detectors are manually positioned and are not moved except for calibration and maintenance.

The intermediate and power channel detectors are designed to operate in the following environment:

1. Gamma flux 10^5 r/hr
2. Thermal neutron flux 1.5×10^{10} $n/cm^2 \cdot sec$
3. Fast neutron flux 1.5×10^{10} $n/cm^2 \cdot sec$
4. Temperature 50 to 150 F.

The start-up channel detectors are fission chambers; the intermediate channel detectors and power channel detectors are compensated and uncompensated ion chambers, respectively. The count rate is presented

on a logarithmic scale; the reactor period is presented in decades per minute. The neutron flux in the intermediate channel is presented on a logarithmic scale while the neutron flux in the power channel is presented on a linear scale.

The nuclear instrumentation system is used to generate certain safety signals and, in addition, supplies information on the neutron flux. The intermediate channel flux level signals are used for the 1 Mw rod withdraw prohibit and the 3 Mw scram during start-up. The neutron flux signals from the power channel are used in the flux-minus-flow scram circuit and in the <17 Mw RWP (Power Range). Figure 7.1.4.2 is a block diagram of the nuclear instrumentation system.

b. Reactor Coolant Instrumentation System

The reactor coolant instrumentation system measures helium flow, temperature, and pressure, and supplies signals to the reactor safety system.

1. Flow

A flow element is located between the outlet of each steam generator and the reactor coolant blower in each of the two reactor coolant loops. The range of flow measurement is from 0 to 110% design flow. The three flow signals from one loop and the three flow signals from the other loop are totalized in pairs to obtain three totalized flow signals. The three total flow signals are compared with the three flux signals from the power range nuclear instruments to give three flux-minus-flow signals for use in the safety system. The flow signals are also used during start-up and in the intermediate range to activate interlocks which insure that coolant flow is greater than 75% of total flow before control rods are withdrawn.

2. Temperature

Two dual-element resistance temperature detectors are located between the reactor vessel coolant outlet and the attemperator in each coolant loop. Each is a fast-response electrically isolated detector. The elements are mounted in thermowells to permit removal of the elements without affecting the integrity of the coolant system. Dual elements for each thermowell are specified so that one of the elements can be classified as a built-in spare which is completely wired to the instrument cabinets located in the main control room; switchover to the spare is manual.

The range of the detectors is from 900 to 1200 F, with a specified accuracy of $\pm 1/2\%$ over this range and a response time of less than nine seconds (at 20% flow); i.e. 63.2% of a step change in the measured variable is indicated within nine seconds. The temperature signals are used in the reactor safety system for alarm and rod withdraw prohibit action at 1065 F outlet coolant temperature and scram at 1075 F.

3. Pressure

Two pressure transmitters are located in each of the two coolant loops between the reactor vessel inlet and the blowers. Three of the four pressure signals are used in the safety system at all times; the fourth pressure signal is completely wired and can be manually switched into the system if one of the three normal signals becomes inoperative. The range of the transmitters is from 100 to

350 psig with an accuracy of $\pm 1\%$ over this range, and with a response time of one second. The pressure signal in the safety system initiates scram actions for both high and low pressure. The reactor coolant low pressure RWP (start-up only) is set at 225 psig with the power and intermediate scram set point at 225 psig; the high pressure alarm is set at 320 psig, with the scram set at 325 psig. The low pressure scram signal also closes the containment shell isolation valves. During start-up the rod withdrawal prohibit signal insures that the reactor coolant system pressure is above 225 psig before control rods are withdrawn.

c. Other Safety System Signals

1. Steam Generator Drum Water Level

Each steam generator drum has six drum water level transmitters. Three transmitters have a response time of two seconds and cover a range of -15 in. to +15 in. from normal water level. The drum level signals initiate a low level alarm at -7 in. and a low level scram at -9 in. The -9 in. signal also actuates contacts which open the electromatic relief valves to blow down steam generator pressure and start all four shutdown feedwater pumps.

The remaining three level transmitters are used to isolate the steam and helium sides of the steam generator from the system. These transmitters measure in the range of 0 to -30 in. from normal water level. Trip contacts at 24-1/2 in. below normal drum water level initiate isolation of both the steam and helium sides of the steam generator.

2. Nozzle Annuli Flow

The reactor vessel design includes provision for nozzle annuli cooling and nozzle interior cooling. A flow element is located in the main header to the nozzle annuli. Three flow transmitters connected across the flow element measure differential pressure and transmit a signal to the reactor safety system. The range of the flow transmitters is from 0 to 15,000 lb/hr helium. The safety system includes a low flow alarm which is set at approximately 7500 lb/hr helium and a low flow scram which is set at approximately 6000 lb/hr helium. A time delay is incorporated into the design to prevent false scrams due to intermittent flow in the system and a scram does not occur immediately upon a low flow signal.

3. Nozzle Interior Flow

A flow element is located in the main header which feeds helium to the interior of the reactor vessel nozzles. Three flow transmitters connected across the flow element measure the differential pressure and transmit a signal to the reactor safety system. The range of the flow transmitter is from 0 to 8000 lb/hr helium. Low flow alarms and delayed scram are specified. The low flow alarm is actuated at a coolant flow of approximately 4000 lb/hr helium and the delayed scram is initiated at a coolant flow of 3000 lb/hr helium.

4. Nozzle Interior Coolant Temperature

Three temperature elements are located in the header which feeds 125 F helium to the service machine and to the interior of the reactor vessel nozzles. Signals from the temperature elements feed alarm, trip, and mixer stages. Signals from the mixer stages are

fed to the safety system. The range of transmitted temperature signal is from 0 to 250 F. The alarm stages are activated at a coolant temperature of 135 F; scram is initiated at a coolant temperature of 150 F.

d. Design Requirements of the Safety System

The reactor safety system is designed to be consistent with the following concepts:

1. Fail-safe

It is impractical, if not impossible, to design a completely fail-safe system. However, fail-safe features are provided where practical. The basic principle applied is that whenever possible, a scram signal is generated in the failed channel due to loss of input signal and/or loss of supply voltage. All components are in the energized condition for normal operation.

2. Reliability

It is desirable to minimize false protective actions which may be caused by safety system component malfunction or failure. To meet reliability requirements, the safety system employs multiple channels, local coincidence circuitry, and operational testing.

3. Redundancy

To increase reliability of the safety system more than one channel of measurement of the same parameter is employed. Except for neutron flux measurement in the start-up and intermediate ranges, at least three measurements are made in all cases. Table 7.1.4.2 shows the number of measurements of each parameter.

TABLE 7.1.4.2

Safety System Measured Parameters

<u>Parameter</u>	<u>No. of Measurements</u>
Flux (int. range)	2
Flux (power range)	3
Reactor outlet temperature	2 per loop - total 4
Reactor pressure	2 per loop - total 4
Reactor flow	3 per loop - total 6
Drum water level	3 per stem generator - total 6
Nozzle annuli flow	3
Nozzle interior flow	3
Nozzle interior coolant temperature	3
Loss of blower (under-current)	3 per blower - total 6
Loss of plant power (under-voltage)	2 per circuit - total 4

4. Coincidence

Multiple signals are used in coincidence circuitry so that failure or malfunction of one channel does not result in plant shutdown. The EGCR employs local coincidence which requires that at least two of the three channels for the same parameter must trip before a scram occurs except for the neutron flux in the start-up and intermediate ranges, the loss of blowers and loss of power. The loss of blower scram occurs when one phase current to each blower is lost or when two phase currents to one blower are lost. Loss of power scram occurs when one phase voltage in each 2.4 KV bus is lost. The safety system measured parameters are listed in Table 7.1.4.2. Each channel includes a trip stage and a mixer stage. The mixer stages receive all three signals from the three channels, and coincidence for each parameter is determined at the mixer stage. The outputs of the mixer stages are fed into three power stages. When two of three power stages are tripped, a scram occurs. Local coincidence is used in preference to general coincidence because the local coincidence type system is less prone to false scrams, and also facilitates system testing.

7.1.5 Rod Control

Twenty-five nozzles located on the reactor vessel top head are provided for the installation of control rods and drive mechanisms. The speed of control rod insertion into and withdrawal from the reactor core was established to prevent exceeding design limitations on the rates of change of temperature of reactor components, piping, and vessels.

a. Regulating Speeds

Control rod speed was determined by analogue computer analyses based upon the criterion that the maximum rate of change of fuel clad temperature shall not exceed 150 F/min.

Three ranges of reactor operation were examined.

1. The start-up range (source to 1 Mw). In this range, there is no appreciable nuclear heat generation and, therefore, no appreciable temperature rise.
2. Intermediate range (1 Mw to 17 Mw). In this range, nuclear heat generation increases and reactor components experience maximum temperature changes.
3. Power range (17 Mw to 85 Mw). In this range, nuclear heat generation is greatest, but temperature changes in reactor components (except oxide fuel temperature) are small compared to the temperature change in the intermediate power range.

Each range was examined⁽¹⁾ to determine the limiting rate at which reactivity could be changed, and plots of both fuel cladding temperatures and outlet gas temperatures were made. In each range a maximum allowable reactivity insertion rate was established corresponding to a fuel cladding temperature change of 150 F/min. From this information, the rod speed of 2 in./min was selected.

b. Scram Actuation

Scram action is accomplished by de-energizing the control rod clutches.

In the safety system three power stages receive all the scram signals. When any two of the three power stages are tripped, contacts in a scram relay matrix open. These contacts remain closed in normal operation and power to the rod clutches is supplied through this circuit. Opening two pairs of relay contacts in the matrix interrupts power to the control rod drive clutches which release the control rod cable drums.

7.1.6 Component Failures

a. Power Supplies

The circuits in the safety system are supplied by three independent power supplies:

1. Panel MG-1
2. Panel MG-2
3. Emergency panel E-5.

Panels (1) and (2) are supplied by individual M-G sets. If one M-G set fails, a manual transfer switch permits transfer to the other M-G set.

The third panel E-5, is supplied from two separate buses of a motor control center with a manual transfer switch installed between the motor control center buses and the panel. Failure of power from one bus of the motor control center results in failure of one channel of instrumentation until the manual transfer is made to the other bus.

Each of the three channels of measurement has its own power supply from one of the three sources. Failure of one power source, therefore, does not cause a scram.

b. Detectors

Electrical detectors can fail due to either open or short circuits. The open circuit is generally easier to cope with since it is indicated as loss of signal. The EGCR safety system is designed so that loss of a signal from the detectors is in the direction of a scram. The flux, pressure, flow, and drum level detectors call for scram action upon loss of signal. The temperature detector, since it is a resistance thermometer, requires a bridge network and is specified to balance the bridge at the high end of the scale (1100 F), when sensor output is zero. Thus, an open circuit appears as an 1100 F temperature, which is above the scram trip point and in the safe direction.

Short circuits are more difficult to cope with. For this reason periodic testing of each channel is required. A short circuit does not generally result in a scram unless it causes a component which is normally energized to trip.

c. Intermediate Components

The intermediate components consist of such items as amplifiers, trip stages, mixer stages, logic circuits, power stages, coils and relays. These items are operated in the energized condition. If power or input signal fails, the component de-energizes and causes a scram trip. Failure of a stage to trip prevents a channel trip and the operator must depend upon alarm annunciation, indicators, or testing to detect and locate the trouble.

7-15

Reactor inlet temperature was selected as the rod controller signal since results of extensive computer analysis had shown that this signal gave optimum control for the disturbances studies.⁽²⁾

The use of reactor inlet temperature to control the rods is subject to the objection that during a cold-coolant accident the control system attempts to withdraw the rods and thus aggravates the accident. However, in the case of the EGCR this objection is not applicable because the cold-coolant accident, in which reactor inlet temperature is suddenly decreased, is not credible.

Operating procedures and electrical interlocks prohibit operation of the EGCR with only one coolant circuit thereby avoiding one method of initiating a cold-coolant accident. With both coolant circuits in operation and the reactor coolant system intact, reactor inlet temperature can be decreased by a decrease in feedwater temperature. Principally because of the large storage capacity of the deaerator, feedwater temperature can only be changed very slowly, with a time constant of approximately ten minutes. Such slow temperature changes are adequately dealt with by the control system.

All other causes of a decrease in reactor inlet temperature are either slow-acting or result in reactor scram, for reasons unrelated to the inlet temperature, before the control system takes any action.

c. Reactor Coolant Blower Control System

Computer studies further showed that signals from the reactor outlet temperature detectors should be used to control the heat removal rate, because they sense heat addition prior to any other sensors and therefore initiate faster correction of any situation in which the fuel is insufficiently cooled.

Reactor coolant flow is so adjusted that reactor outlet coolant temperature is maintained at its desired value as reactor power changes. Coolant flow is varied by changing the speed of the reactor coolant blowers. The average outlet temperature from the reactor is combined with a flow signal from each loop; the two resulting signals are fed to the individual blower speed controllers. The error signal produced when the incoming signal is compared to the set point causes a change in blower speed. To synchronize the speed of the two blowers, an additional signal from a tachometer on each blower is supplied to the controller of the slower of the two blowers; the controller will speed up the slower unit to match the speed of the other blower. The equation for the blower controller, not including the speed signal, is as follows:

$$E = a \left[K_p e_T + K_R \int_0^t e_T dt + K_D \frac{de_T}{dt} \right] - b \left[W_{g_{set}} - W_g \right] \quad \text{Eq. (7.2.3.1)}$$

where:

E = controller error signal

a = gain-sensitivity adjustment

b = gain-sensitivity adjustment

K_p = proportional gain

K_R = reset gain

where:

e = error signal to controller

W_f = the feed flow

W_s = the steam flow

L_S = the drum level set point

L_a = the actual drum level

$\left. \begin{matrix} K_f \\ K_L \end{matrix} \right\} =$ characteristics of the controller,
representing the feedwater flow and
drum level gains.

The first part of the equation represents the portion of the error signal generated by a mismatch between steam and feedwater flow. The second part of the equation represents the portion of the signal generated by a mismatch between the actual drum water level and the desired level. It is desirable to have the feedwater controller level-dominant, because any error in flow measurement causes an error in steam drum water level. The drum level error due to flow measurement error is a direct function of K_f/K_L .

Figure 7.2.3.3 is a diagram of the feedwater level control system.

f. Steam Dump System

The steam dump system acts as a second heat sink when the turbine does not require all steam generated. Steam is taken from the steam main before the turbine stop valve and is regulated by a dump control valve. The dump flow passes through two desuperheaters into the main condenser. The dump valve controller setting is 1280 psig which is 30 psi above the turbine initial pressure setting of 1250 psig. The dump valve goes from fully closed to fully open over a pressure range of 15 psi, or from 1280 psig to 1295 psig. Any increase in initial turbine pressure which cannot be controlled by the turbine governor system opens the dump valve to bypass steam to the main condenser. This bypass flow reduces the initial pressure at the turbine.

A load limiting device which sets the maximum opening of the turbine governor valve determines the maximum steam flow to the turbine. Any excess steam generated is passed to the steam dump.

7.2.4 Control System Components

a. Primary Elements

The primary elements in the control system consist of temperature, flow, level, and pressure detectors.

The temperature elements for both the rod controller and the blower controller are dual element, fast response, resistance type temperature detectors. They are installed in thermowells, to permit removal without disturbing the integrity of the coolant system. The range of the temperature detectors for the rod controller is from 350 to 550 F; the range of the detectors for the blower controller is from 900 to 1100 F. The response of the detector is less than nine seconds (20% flow) to measure 63.2% of a step change in the measured variable. ←

The flow element which supplies signals to the blower control system are Dall-tubes located in the blower inlet coolant piping. Differential pressure transmitters, located across each tube sense the differential pressure and transmit these signals to the blower control network. The transmitters cover a range of 0 to 110% of design flow. The response of the transmitters is two seconds to measure 63.2% of a step change in the measured variable.

The level transmitters which measure steam drum water level are differential pressure type electronic transmitters which measure the difference in head between a constant water column and the actual water level in the steam drum. The range of measurement is ± 15 inches of water level. The response of the detectors is two seconds to measure 63.2% of a step change in the measured variable.

A steam pressure signal is used for turbine control and steam dump control. The turbine controller is an initial pressure regulator which has 2-3% regulation. It can maintain a steam pressure of 1250 psig at 100% flow and 1215 psig at 20% flow. The pressure transmitter and controller for the steam dump system operates over a range of 250 to 1350 psia.

b. Intermediate Components

The intermediate components consist of such items as amplifiers, summing and averaging circuits, and controllers. The components are standard manufactured items.

Amplifiers are used to amplify the detector or transmitter signals and to convert these signals into usable form consistent with the remainder of the circuitry.

The averaging circuits are used in the rod control and blower control systems. The reactor inlet coolant temperatures are averaged to form a signal to actuate the rod controller. The outlet temperatures are also averaged to produce a signal to actuate the temperature controller which in turn actuates the blower controllers.

Controllers are provided with proportional, reset and derivative gain adjustments. Each controller is equipped with a recorder, set point indicator, and manual-automatic transfer switch to permit bumpless transfer from one mode of control to the other.

7.2.5 Main Control Room, General Description

The main control center is designed to function as the principal operating station for the over-all plant, with the exception of the service machine, charge machine, and burst-slug detection system equipment. Measurements from primary elements located throughout the reactor building and turbine building are transmitted to the main control room where the remote operating controls and control information are centralized and displayed.

a. Control Room Layout

The main control room is located on the north side of the ground floor of the control building. The control panels, desks, and switchramps are arranged with instruments and controls associated with the reactor core temperature and radiation monitoring on the left side when facing north. Continuing around the control room to the right are the reactor section, helium section, steam section, and the electrical section on the far right. The principal operating panel boards, control desks, and

auxiliary equipment panels in the main control room area are listed in Table 7.2.5.

TABLE 7.2.5

Main Control Room Equipment

	<u>Title</u>	<u>Panel No</u>
→	PTM system monitoring cabinet	KS-51-5
	Radiation monitoring control desk	K-1-1
	Radiation monitoring and temperature monitoring duplex panel and auxiliary relay rack	K-1-5 and K-1-5a
	Reactor duplex panel (with integral switchramp K-1-2) and auxiliary relay rack	K-1-6 and K-1-6a
	Helium panel (with integral switchramp K-1-3) and auxiliary relay rack	K-1-7 and K-1-7a
	Steam duplex panel (with integral switchramp K-1-4) and auxiliary relay rack	K-1-8 and K-1-8a
	Electrical control desk	
	Electrical duplex switchboard	N-6
	Failure-free power instrument panelboard (MG-1)	N-86
	Failure-free power instrument panelboard (MG-2)	N-87
	Emergency power instrument panelboard (E-5)	N-149
→	Instrument voltage stabilizing transformer	N-157
	Emergency instrument panelboard (E-1)	N-145
	Carbon dioxide control panel	N-156
	Electrode relay cabinet	LC-21-1
→	Terminal cabinet (telephone)	
	Code transmitter	
→	Emergency cooling loop control panel	
→	Fire and evacuation alarm equipment	
	Building radiation monitoring cabinet	RI-20-1B
	Lighting panel A	
	Main power transformer tap changer control panel	N-15
	Fire and supervisory annunciator panel	

The desk of the shift supervisor and the operator's desks face the main control panels.

Access to the main control room is by the door off the east-west

2. Failure of control channels and equipment results in control action in the safe direction.

- i. Accuracy

Requirements for accuracy are uniformly specified as the allowable deviation of the reading, action or signal from the actual steady-state value expressed as a percentage of the measured, indicated, or controlled range.

- j. Control Stability

Where control system designs are new and unique, analogue simulation techniques were used to establish system stability. Instrumentation designs for the simpler, more conventional systems were based on experience and similarity with existing systems.

- k. Standard Products

Standard commercial instruments of proven operability and reliability are used in all applications where such instruments are available.

- l. Compliance with Performance Requirements

Control over the equipment which is supplied, and assurance that specification requirements are met, is obtained through the following steps:

1. Review and approval of vendor drawings
2. Independent inspection, tests, and review of certified test reports for work performed in the vendor's shop prior to fabrication or shipment
3. Inspection and calibration tests performed on receipt at the plant site prior to installation
4. Performance and acceptance testing during initial plant start-up. Vendor representatives are to assist in alignment of instrumentation and correction of any deficient items.

- m. Reactor Operations Supervision

Plant operating objectives and safety objectives require that limitations on permissible operator actions be maintained during all phases of operation. Continuous supervisory control of all phases of the plant operation is necessary since provision in the design of adequate interlocks and preventives against all improper operator actions is impractical. ←

With respect to instrumentation, supervisory control will include but not be limited to testing, calibration, repair, and alignment of plant components or instruments as part of a planned and scheduled program. ←

7.3.2 Temperature, Pressure, Flow, and Level Sensors

The four process parameters measured for plant control are temperature, pressure, flow, and level.

The types of temperature sensors used are thermocouples, resistance thermometers, and bimetallic elements. Temperatures throughout the plant are measured in degrees Fahrenheit. Wells for test calibration and adjustment of sensors are provided.

The pressure sensing elements are Bourdon tubes, diaphragms or bellows, and transducers. Instrumentation connection lines for all pressure and differential pressure transmitters are designed to permit isolation of the detector from the plant processes and to permit insertion of a test pressure for calibration purposes. Removal and replacement of defective units can be accomplished without depressurization of process lines.

Flow metering of fluids in closed pipes may be accomplished by measuring the pressure differential across a suitable flow restriction. The flow sensors which are used include orifices, Venturis, Dall tubes, and rotometers. Flow measurements in which the fluid temperature or pressure varies over an appreciable range require compensation for the corresponding changes in fluid density and absolute viscosity; where necessary, such compensation is provided.

The techniques for measurement of liquid level or static head employ gage glasses, differential pressure head meters, and positive displacement float meters.

7.3.3 Special Instrumentation

A variety of special purpose instrumentation is used as given in Table 7.3.3.

TABLE 7.3.3

Special Purpose Instrumentation

<u>Application</u>	<u>Instrument Type</u>
Analyzers	
Moisture content	Electrolytic hygrometer
Helium contaminants	Chromatographic analyzer Hydrocarbon analyzer
Steam quality	Calorimeter
Steam condensate and feedwater	Conductivity meter Hydrogen ion pH meter
Vibration	Accelerometer
Rotational speed	Tachometer
Position	Servo mechanism Differential transformer Mechanical measurement
Helium detection	Thermal conductivity meter
Activity detectors	
Coolant loop gross activity. .	Gas quenched Geiger-Mueller counter (gamma detector)

collecting tape is presented to a scintillation counter located outside the precipitation chamber. The counter measures the radioactivity present on the tape and hence the activity present in the reactor fuel channel. A rise in activity beyond the expected statistical fluctuations of background activity is an indication of the presence of a ruptured fuel element in the one of eight fuel channels from which the composite sample was taken. The eight channels are then individually monitored to locate the particular channel containing the ruptured fuel element.

The ions produced and the activity measured both result from the beta decay of fission-product decay chains.

The detection and readout instrumentation sub-system is made up of three identical instrumentation channels (one each to satisfy the counting and recording needs of each monitor), the programming equipment, the selector-valve "on-line" indicators, and the main control room BSD recording and annunciator equipment. Other than the readout equipment supplied for the main control room, all detection and readout instrumentation is located in the BSD instrumentation room located on the third floor of the reactor building.

Each instrument channel consists of the following instrument stages:

- a. Monitoring (precipitation and detection)
- b. Amplification
- c. Discrimination
- d. Counting
- e. Recording
- f. Digital-to-analogue conversion
- g. High-count decoding
- h. High-count annunciation.

Signal pulses derived from the scintillation detector are amplified and sorted as to energy (voltage) level and fed to a decade scaler for counting over a preset time interval. The total count is printed and an analogue voltage proportional to count is generated. If this voltage exceeds preset limits, the high-count decoder actuates the high-count annunciators.

The gas-handling sub-system conveys gas samples to detection system instrumentation and returns the gas to the reactor coolant system. A gas-sample line extends from each core fuel channel to the BSD equipment room located at the east face of the biological shield. In the course of the run of tubing, four special penetrations are made: two in the wall of the pressure vessel (117 lines each) and two in the biological shield (117 lines each). The latter two also serve as heat exchangers to cool the sample gas entering the monitors.

Upon entering the BSD equipment room, the 234 lines are fed directly to 30 block-manifold valves each accommodating eight lines. Each block serves as a housing for 24 valves: eight manually-operated, normally-closed individual sampling valves; eight manually-operated, normally-open isolation valves; and eight normally-closed solenoid valves which lead to the electro-pneumatic selector valve. The manifold valves are programmed to permit composite samples from eight lines to be fed into a flow-direction valve arrangement and then to the electro-pneumatic selector valves. Flow direction valves channel the sample flow to either a primary or an auxiliary bank of 30 electro-pneumatic selector valves each of which feed the primary or auxiliary detector unit. The normally-closed electro-pneumatic selector valves are sequentially opened to feed composite gas samples to the monitor. The eight normally closed valves are operated manually in the BSD equipment room to determine which channel out of the group of eight contains the failed fuel.

Three monitors are installed for normal (primary monitor), location (location monitor), and spare monitoring (auxiliary monitor). An inter-connecting piping and valve arrangement upstream of the monitors permits each unit to perform any of the three monitoring functions. The gas samples are filtered prior to entering the monitors. From each monitor the gas passes through flow-control equipment to a suction vessel. The suction vessel has two outlets. The first outlet is a pressure-relief arrangement which discharges to the gas vent system. The second outlet is to the BSD compressor (one operating and one stand-by) which returns the sample gas to the reactor coolant system.

A supplemental gas-handling system furnishes clean helium from the plant make-up system for the scintillation phosphor purge. The system consists of three manually-operated flow-control valves (one for each monitor). Purge gas into the monitors returns to the reactor coolant system along with the BSD sample gas flows.

The BSD instrumentation installed in the main control room consists of three high-count annunciators (one for each instrument channel), a system-malfunction annunciator, and a strip-chart recorder to record the analogue voltage proportional to count. The recorder is supplied with a selector switch, to permit recording the output of any of the three instrument channels.

The two programmers (one operating, one spare) coordinate the operation of the gas-sampling and the detection and readout instrumentation of the BSD system.

The gas-sampling processes of the BSD and PTM (Section 7.4.5) systems are operated by separate programmers, which are interlocked to prevent interference of the two systems.

7.4.5 Pneumatic Temperature Monitoring System(4)

The pneumatic temperature monitoring (PTM) system is used to measure outlet gas temperature of individual fuel channels during steady-state reactor operation. The PTM system provides a backup for the thermocouple system (Section 7.4.2). The thermocouples are not replaceable and are not expected to last the life of the reactor, whereas the PTM system affords a permanent and reliable method of monitoring the fuel channel outlet temperatures for the life of the reactor. All of the PTM components except for the primary nozzles are outside the reactor pressure vessel.

The PTM system uses the available BSD system sampling tubes (Section 7.4.4) from the reactor core fuel channel extensions to the BSD valve manifolds. Primary sonic nozzles are installed in each of the BSD tubing connections to the core fuel channel extensions. Two secondary sonic nozzles, instrumentation and programming equipment, are located in the BSD equipment and instrumentation rooms. This equipment permits sequential measurement of fuel channel outlet gas temperatures. The scanning times of the PTM and BSD systems are programmed to preclude interference between the two systems.

The operation of the pneumatic temperature probe is based on the fact that sonic flow of a gas through a nozzle is a function of the temperature of the gas entering the nozzle:

$$W = \frac{CAP}{\sqrt{T}}$$

Eq. (7.4.5.1)

where:

$$C = \sqrt{\frac{K}{R} g_c \left(\frac{2}{K+1} \right) \frac{K+1}{K-1}}$$

six or less can be used simultaneously. In-core guide tubes are permanently installed in the reactor core, and leave the reactor vessel through the bottom thermocouple nozzles. The design of the guide tubes permits the use of the following detector types:

- a. Activated wire detectors
- b. Ion chambers
- c. Thermopiles.

Activation wire detectors and ion chambers are used in the EGCR to measure reactor core thermal neutron flux from 10^9 to 10^{14} n/cm²·sec. This range corresponds to approximately 0.001 to 150% of design reactor power.

Wires, inserted into the guide tubes which extend the full length of the core, are allowed to soak for a predetermined time and are then withdrawn and passed through a counter to determine the amount of induced activity in the wire.

The wires are positioned by drive units located exterior to the biological shield. Each drive unit is controlled by an individual drive switch which enables the operator to drive the wires to an indexing point. When all six detectors are positioned at the indexing point, a master control switch is used to drive all wires into the core at a fast speed. Each drive unit is arranged so that the activated wire can be inserted into any one of three guide tubes. In this manner six channels are measured, counted and recorded. After a specified decay period, the wires can be inserted in six other channels to obtain additional measurements.

Each drive unit is designed to store wires when not in use. The switching from one guide tube to another is done through solenoid transfer devices. Limit switches automatically prevent excessive wire withdrawal and insertion. Panel lights display the position of the wire in transit from the storage unit into the core and also from the core to the counter.

Three 2π -geometry scintillation crystal counters permit the simultaneous reading of three wires. Each wire is passed through a counter in discrete steps and a count per step per unit time is obtained to give wire activation at each section of wire. This activation is recorded on punched tape for processing, interpretation and final use by the reactor operator to determine core flux profiles. The ion chambers or wire may be used during high power operation. The output of the chambers is fed to flux amplifiers and recorded on strip chart recorders.

7.5 Process and Building Radiation Monitoring

7.5.1 Philosophy

The radiation monitoring system is designed to give a continuing indication of radiation levels within the plant and its controlled area and to monitor releases of radioactivity to the environment outside the EGCR control area. In general, the localized presence of high-level radiation within the plant is not a cause for automatic containment shell isolation, but annunciation is provided to alert reactor operating personnel who can manually perform the necessary corrective or emergency actions.

Monitors for air-borne radioactive particulates (continuous air monitors) and area monitors (gamma radiation) are installed for protection of plant personnel and as indicators of possible reactor equipment malfunction. Local indication of activity level and audible alarms are included, where necessary, for protection of personnel adjacent to potential sources of radioactivity. In

→ addition, most instruments are connected to recorders and annunciation alarms in the main control room or in the service and charge machine control room.

Portal monitoring systems and hand and foot counters are located at exits from possible contamination areas to normally clean areas. These units are equipped with local indicators and alarms.

The stack effluent is continuously monitored for radioactive particulates, gaseous fission products, and radioiodine. When total gaseous activity exceeds a predetermined allowable level, the containment shell is isolated automatically. Other monitored activity levels provide information to the operator so that procedures to shut down the reactor can be initiated.

→ Radioactive liquid wastes, which are generated in relatively small volumes, are monitored before transfer from the reactor building to either the high-level waste storage tank or to the low-level retention basin. After analysis, the low-level waste is diluted with condenser cooling water, monitored, and discharged to the Melton Hill Lake. A water monitoring station has been established at Melton Hill Dam, and a program of mud, liquid, and biota sampling is included in the monitoring program.

The existing ORNL perimeter monitoring system,⁽⁵⁾ with an additional station, will be used for determination of environmental contamination in the immediate plant area and the area on the opposite bank of Melton Hill Lake. An additional air monitoring station is located south of Melton Hill Lake on TVA land adjoining the lake and is connected to the existing system.

7.5.2 Gaseous Waste Monitoring

The gaseous waste monitoring system consists of two stack monitors, a gross activity monitor on the helium sampling lines from the service machine and the charge machine and a gross activity monitor on the helium vent lines from the charge and service machines (Section 6.3).

The function of the stack monitors is to obtain information regarding radioactivity release to the atmosphere and to initiate automatic closure of the containment shell isolation valves in case of excessive gaseous activity level. The activity monitor on the sampling line from the charge or service machine is used to determine the disposition of helium which is vented from the charge or the service machine and stops helium transfer if high coolant activity is detected during transfer.

The stack monitor samples the activity in any discharge from the gas vent system and from ventilating air discharged from the service machine room, steam generator rooms, charge machine room, and service building. Biological shield cooling air is also discharged to the stack and is monitored.

The helium vent system accommodates gas which is vented from relief valves, hand control valves, and vent lines from the following systems and components:

- a. Reactor pressure vessel
- b. Steam generators
- c. Gas purification and recovery system
- d. Evacuation and helium transfer system
- e. Burst slug detection system
- f. Service machine
- g. Charge machine
- h. Fuel loading and rehearsal mechanism
- i. Nozzle and service machine coolers
- j. Plant air equipment

Each of the two stack monitoring systems consists of an isokinetic sampling nozzle which penetrates into the gas stream in the stack, a particulate matter activity monitor, two total gas activity monitors, an iodine monitor, valves, flowmeters and two ten-cfm blowers. Sample gas is pulled from the stack and passes through the moving filter sampler of the particulate matter activity monitor, which is fitted with a beta scintillation detector. A rate meter and recorder for this monitor is installed in the main control room.

The discharge from the particulate matter activity monitor flows in series through the two total gas activity monitors. Each of the total gas activity monitors has a gamma scintillation detector (with check source mechanism) which transmits to remote readouts in the main control room.

Approximately 90% of the discharge of the two total gas activity monitors goes directly through a flowmeter and the blower for return to the stack and approximately 10% of the discharge is bypassed to the iodine monitor prior to return to the stack. The iodine monitor includes a charcoal trap (with a replaceable cartridge), and a gamma scintillation detector in a lead shield. The readout and recorder are located in the main control room.

The detectors on the charge and service machine vent and sampling lines read out and alarm in the service and charge machine control room. The detectors are halogen-quenched Geiger-Mueller tubes.

7.5.3 Reactor Coolant System Monitoring

The monitoring system for the reactor coolant includes a radiation detector on each reactor coolant circuit located on the steam generator discharge line. An alarm condition is indicated by manually reset lights in the main control room.

The detectors are mounted within three feet of the centerline of the coolant piping. They are halogen-quenched Geiger-Mueller counters with a built-in solenoid operated check source.

7.5.4 Liquid Waste Monitoring

The liquid waste monitoring system consists of monitors used to determine the activity of liquids which are to be discharged into the liquid waste system. The liquid waste system provides for the disposal of hot wastes by tank trucks to the ORNL liquid waste disposal facilities and the eventual disposal of warm wastes into the Clinch River.

The largest single volume of liquid which may require disposal is the water from the spent fuel storage basin. A radiation monitoring system consisting of a pump, demineralizer, filter and detector is installed so that either the special spent fuel monitoring tanks (in which spent fuel elements are checked for leaks prior to storage) or the fuel discharge chute or basin water may be monitored for contamination. The gamma scintillation type detector senses integrated activity in the demineralizer anion column which follows, in series, a glass wool filter and a cation column. The recorder is locally mounted on an instrument panel.

Other monitors in the system of liquid waste disposal are on the discharge lines of the warm and hot waste sump tanks, the service water return line (to the service water return sump), discharge from the blower seal water coolers, and discharge from the blower seal water demineralizers. The above monitors all read out, annunciate, or record in the main control room. The detectors are of the gamma scintillation type.

7.5.5 Building Direct Radiation Monitoring

The direct radiation monitoring system serves to monitor all accessible areas for the presence of gamma radiation. The detectors of this system are located to afford maximum coverage and remote readout is located either in the main control room or in the charge and service machine control room, depending on the particular areas or equipment monitored. The detectors in normal work areas include local alarms. A local monitor is installed on the traveling bridge of the spent fuel storage basin. Also included in this system are the personnel monitoring probes located throughout the plant at various exits from areas where contamination may occur such as entries to stairwells, corridors and doorways. Four hand and foot monitors with alarm buzzers and lights are located at strategic locations throughout the plant.

A twenty point recorder, in the main control room, receives signals from 19 Geiger-Mueller detectors located as follows:

→ a. Service and turbine building

-
1. Service aisle, south side
 2. New fuel storage vault
 3. Spent fuel storage basin area, north side
 4. Spent fuel storage basin area, south side.

b. Reactor building

1. Evacuation and helium transfer equipment room
 2. Basement No 1
 3. Reactor coolant gas treatment equipment room, north end
 4. Reactor coolant gas treatment equipment room, south end
 5. Heating and ventilating equipment room
 6. Ground floor near the personnel entrance
 7. Ground floor near the elevator
 8. Ground floor over the future experimental loop location, west side of the reactor
 9. Ground floor over the future experimental loop location, south side of the reactor
 10. Burst slug detection system pipe room
 11. Third floor adjacent to the future experimental loop location, north side of the reactor
 12. Third floor adjacent to the future experimental loop location, west side of the reactor
 13. Third floor adjacent to the future experimental loop location, south side of the reactor
 14. Fourth floor, biological shield coolant loops
 15. Flux scanner drive motor room.
-

c. Control building main control room, north side

Local audible alarms are included except for the main control room detector and the detectors for future experimental loops.

A second twenty point indicating recorder in the main control room receives signals from 12 Geiger-Mueller detectors located in the reactor building as follows:

-
- a. Charge machine room, west end
 - b. Charge machine room, south end
 - c. Service machine room, south wall
 - d. Service machine room, west end
 - e. Fifth floor, exhaust equipment room

- f. Fifth floor, heating and ventilating equipment room
- g. Sixth floor, north side
- h. Sixth floor, south side
- i. Blower motor room No 1
- j. Blower motor room No 2
- k. Third floor, steam generator No 1
- l. Third floor, steam generator No 2.

Audible local alarms are installed with the detectors in the charge machine, service machine and reactor coolant blower motor rooms.

Nine indicators receiving signals from Geiger-Mueller detectors associated with charge or service machine operation are located in the charge and service machine control room. The detectors are located in the reactor building as follows:

- a. Second floor, control room
- b. Second floor, fuel loading room
- c. Ground floor, charge machine rehearsal room
- d. Third floor, service machine rehearsal room
- e. Charge machine room, south wall
- f. Charge machine room, north wall
- g. Service machine room, north wall
- h. Reactor coolant gas treatment equipment room⁽²⁾.

Local audible alarms are installed for the detectors in the charge machine room and the service machine room.

7.5.6 Building Air Monitoring

The building air monitoring system consists of eight fixed air filter particulate matter monitors located throughout the EGCR buildings with installed wiring for the further addition of eight which, if required, can be placed as determined by operating experience. A 10-point multipoint recorder is installed in the main control room with spare panel space for the installation of a duplicate recorder. The recorder has an adjustable set point alarm. Each detector includes a local alarm and rate meter. Inlet and outlet sample hoses, 8 ft long, are used with two of the detectors which are adjacent to the charge machine room and service machine room. The locations of the eight detectors are as follows:

- a. Reactor building
 - 1. Reactor coolant gas treatment room
 - 2. Adjacent to the charge machine room, basement No 2
 - 3. Ground floor on the east side of the reactor
 - 4. Burst slug detection equipment room, third floor
 - 5. Adjacent to the service machine room, fourth floor
 - 6. Second floor, outside of the service and charge machine control room.
- b. Service building - adjacent to the spent fuel storage basin, north wall.
- c. Control building - main control room.

Additional wiring is installed for future detectors which may be required in the following areas:

- a. Service building
 - 1. The corridor outside of the personnel air lock

2. The corridor outside of the equipment air lock.

b. Turbine building

1. Mezzanine floors, H & V equipment area
2. Second floor, H & V equipment room

c. Reactor building

1. Evacuation and helium transfer equipment area
2. Biological shield cooling equipment area
3. Service machine equipment room
4. Fifth floor elevator corridor to the exhaust equipment area.

7.5.7 Perimeter Air Monitoring

→ Nine perimeter air monitoring stations are located within a radius of 7 miles from the EGCR site. All of the stations have been installed and are used by ORNL in conjunction with laboratory operations.

→ Information from all stations is telemetered via land-line or radio facilities to a central location in the ORNL complex. The same information is re-transmitted from the ORNL monitoring center to the main control room at the EGCR.

Under normal operating conditions, station data is recorded sequentially on multipoint strip chart instruments. The system design permits continuous monitoring of a particular station if this is necessary or desirable; sequential monitoring cannot be done at the same time. It is possible for either operator (at the EGCR or at the central monitoring point at ORNL) to select a particular station for special monitoring and data recording but two separate stations cannot be monitored simultaneously.

- c. Type and purity of graphite
- d. Partial pressure of oxygen
- e. Prior neutron irradiation of the graphite
- f. Gamma radiation during oxidation.

An experimental program has been conducted by Hanford Atomic Products Operation to determine the rate of oxidation of EGCR type graphite. The experimentally determined rate in the absence of gamma radiation is:

$$k = 6.91 (10^{10}) \exp \frac{-90,000}{RT} \quad \text{Eq. (8.4.5.1)}$$

where:

k = reaction rate, mass fraction reacted per hour
 R = universal gas constant
 T = temperature, R

The graphite oxidation rate in the presence of gamma radiation of 10^7 r/hr intensity expected immediately following shutdown of the EGCR, had been estimated to be: (14)

$$k = 8.33 (10^3) \exp \frac{-36,000}{RT} \quad \text{Eq. (8.4.5.2)}$$

For this rate the temperature above which gamma radiation does not significantly affect the graphite oxidation rate is 675 C. However, the oxidation rate in a 6.6×10^6 r/hr gamma intensity was found to be not significantly different from the rate in Eq. 8.4.5.1 down to 450°C.

The graphite oxidation rates given by equations 8.4.5.1 and 8.4.5.2 were used for the hazards analysis.(13)(46) In addition, factors were used to modify these rates to account for such things as graphite purity and neutron irradiation.(14) The results of this analysis are discussed in Section 8.10.5.

8.4.6 Graphite and Water Vapor

The reaction between graphite and water vapor,

is endothermic and, consequently, the reaction itself is of no concern as a potential source of chemical energy. However, the subsequent reaction between hydrogen and oxygen,

is exothermic and can occur with explosive violence if the hydrogen concentration is within 3.99 to 74.2% by volume. The production of hydrogen can occur following several accident sequences. This section presents only the reaction rate data for graphite and water vapor. Discussions on hydrogen generation for specific accidents are presented in the sections describing the accidents.

The rate of reaction between nuclear grade graphite and steam at 300 psia is given in Table 8.4.6. The reaction rate is affected by the system pressure. At one atmosphere pressure the reaction rate is approximately 1/14 of the rates

given in Table 8.4.6.

TABLE 8.4.6⁽¹⁵⁾

Graphite Steam Reaction Rates (300 psia)

<u>Temperature (F)</u>	<u>Graphite Reaction Rate (g/ft²·hr)</u>
800	0.1
1000	1.0
1200	7.2

The reaction rate between graphite and steam is reduced by the accumulation of hydrogen in the system. For example, if the concentration of hydrogen and steam are equal, the rate is 20% of that shown in the table. If the hydrogen concentration were to exceed the steam concentration by a factor of 10, the reaction rate would be 10% of that given in the table.

The rate of reaction is not significantly enhanced by gamma radiation. Experimental evidence from experiments conducted at the U. S. Bureau of Mines⁽¹⁵⁾ indicates less than 1% increase in the reaction rate at 1000 and 1500 F under a gamma flux of 10^7 r/hr.

8.4.7 Graphite and Hydrogen

Hydrogen in quantities sufficient to react significantly with graphite could only result from large amounts of steam being admitted and reacting with graphite to produce hydrogen as follows:

The admission of steam will be detected almost immediately by instrumentation in the reactor coolant system, a high moisture alarm will sound, and the reactor would be shut down by the operator or automatically by a high coolant pressure scram. In the meantime, hydrogen would react with the moderator graphite to produce small amounts of methane. While this reaction is slightly exothermic ($\Delta H = -17.9$ kcal/g-mole), the reaction rate is low at the temperatures of interest so that no significant energy release would result.

Experiments in BEPO have shown that radiation accelerates the reaction of hydrogen and graphite and that the radiation-induced reaction has a much smaller temperature dependence than the thermal reaction.⁽¹⁶⁾ At 500 C the in-pile reaction at a pressure of 800 mm Hg is approximately 10 times greater than the thermal reaction. However, at 600 C the corresponding ratio is 4.4 and at higher temperatures the difference would be even smaller. At 800 C the methane concentration due to thermal reaction is on the order of 4%. Thus it is apparent that some methane would be produced; however, no evidence has been found of mass transfer of carbon by decomposition of methane under these conditions.

8.4.8 Graphite and Carbon Dioxide

The generation of CO₂ in the reactor coolant could occur from the reaction of steam with graphite, as discussed in the previous section, or from the oxidation of graphite by air admitted through a rupture in a coolant line.

The reaction of steam with graphite produces CO and CO₂ as well as hydrogen. The CO₂ may in turn react with graphite, producing more CO, and the reaction

TABLE 8.5.2

Estimate of Stored Energy in EGCR Graphite
(After 2 years of Reactor Operation)

<u>Irradiation Temperature (c)</u>	<u>Total Stored Energy Based on Annealing Out of Pile at 1000 C for 3 hours (cal/g)</u>	<u>Comments</u>
250	50	Based on Figure 7 of reference (26), a value less than 10 cal/g is obtained for 1135 Mwd/AT exposure. This exposure is about one-fifth the expected exposure in the EGCR after 2 years. Making the conservative assumption that stored energy is directly proportional to exposure, the value of 50 cal/g is obtained. This reference does not state at what temperature the samples were annealed or the type graphite used.
400	14	Based on reference (27) for exposure approximately equivalent to 2 years of EGCR operation. CSF graphite was used.
750	9	Based on reference (27) for exposure approximately equivalent to 2 years of EGCR operation. CSF graphite was used.

8.6 Reactivity Accidents

8.6.1 General

Uncontrolled reactivity insertions can occur as a result of a rod runaway, by improper handling of fuel or control rods during the refueling operation, or by the addition of water or steam into the core.

Uncontrolled reactivity insertions as a result of a cold coolant accident cannot occur in the EGCR. The safety system prevents reactor start-up unless both blowers are operating and the flow is at least 75% of a full flow. If either blower trips at any time during reactor operation, the reactor will scram. Therefore, it is not possible to cause a cold coolant accident by restarting a reactor coolant blower. Furthermore, interlocks prevent operating individual isolation valves when the blowers are operating. This precludes the possibility of causing a cold coolant accident by first isolating and then reopening a coolant circuit.

Spatial oscillation of the power distribution of the core due to xenon cannot occur. The relatively small core and the small temperature coefficients associated with the EGCR are below the threshold values for such oscillations. A temperature coefficient of reactivity for the moderator of at least $53 (10^{-5})/C$ is required to induce oscillations in the axial direction. To induce oscillations in the radial direction the moderator coefficient must be at least $72 (10^{-5})/C$. (32)

Uncontrolled reactivity insertions due to rod runaway were studied on an analogue computer. The following cases were considered:

- a. Withdrawal of ten control rods in the power range
- b. Withdrawal of two control rods in the intermediate range
- c. Withdrawal of thirteen control rods in the source range (start-up accident).

These represent the maximum permissible numbers of rods that can be moved simultaneously plus one additional rod. If more than the permissible number of rods are selected, a rod withdraw prohibit prevents withdrawal of any rods. However, a single failure in the rod withdraw prohibit circuit could allow one additional rod to be withdrawn.

8.6.2 Rod Withdrawal at Power

The power range extends from 17 to 85 Mw. In this range, a maximum of nine control rods can be moved to control reactor flux. The rod runaway accident was simulated for initial power levels of 17 and 85 Mw with an assumed failure in the rod limiting circuit.

The coolant flow was set to correspond to the initial power level. For example, 20% flow corresponds to a reactor power of 17 Mw and 100% flow corresponds to a reactor power of 85 Mw. In both cases, k_{eff} was initially unity. Ten control rods were withdrawn at the design rate of two inches per minute in the region of maximum worth.

Table 8.6.2 shows the maximum reactor power, cladding temperature, and reactor outlet temperature for power range runaways from initial power levels of 17 and 85 Mw. A flux minus flow scram is effective in maintaining fuel cladding and reactor outlet temperature within the operating limitations. Thus, power range runaway does not cause a hazard.

In the intermediate and source ranges the transient is self limiting and no safety system action is required. Maximum fuel cladding temperature, reactor outlet coolant temperature, and reactor power are shown in Tables 8.6.5.1 and 8.6.5.2.

TABLE 8.6.5.1

Step Insertion of 0.1% Excess Reactivity

	<u>Source Range</u>	<u>Intermediate Range (1Mw)</u>
Maximum reactor power	9 Mw	10.3 Mw
Maximum reactor outlet coolant temperature	511 F	539 F
Maximum fuel cladding temperature	560 F	591 F

TABLE 8.6.5.2

Step Insertion of 0.1% Excess Reactivity In Power Range

<u>Initial Power</u>	<u>17 Mw</u>		<u>85 Mw</u>	
<u>Blower Control</u>	<u>Manual</u>	<u>Automatic</u>	<u>Manual</u>	<u>Automatic</u>
Maximum reactor power	29 Mw	29 Mw	105 Mw	107 Mw
Maximum reactor outlet coolant temperature	985 F	990 F	1051 F	1043 F
Maximum fuel cladding temperature	1205 F	1205 F	1438 F	1438 F

8.6.6 Increased Reactivity from Water or Steam Addition to the Core

The addition of water into the reactor core increases the reactivity. Complete flooding of the core would increase the multiplication factor by approximately 0.06. Complete flooding of the reactor core is highly improbable, and cannot occur as a result of a steam tube rupture since the bottom of the steam generator would have to be filled before any water could reach the reactor vessel and water would be shut off automatically to the steam generator containing the failure and that steam generator isolated before water fills the bottom of the steam generator.

Flooding from external water supplies is also unlikely. Approximately three million gallons of water would be required to flood the containment shell up to the level of the reactor vessel. Sufficient time is available for any leak to be discovered and repaired before the containment shell could be flooded.

As discussed in Section 4.7.2, the shutdown worth of the full 21-rod bank is $0.241 \Delta k$. The largest k_{eff} that an EGCR core could have is 1.161 in the cold clean initial core. Thus, if water were to completely flood the reactor core the maximum k_{eff} could be 1.23. If one control rod had been removed from the reactor at the time flooding occurred, but with the remaining 20 control rods fully inserted, the value of k_{eff} would be 0.99. If an additional control rod were unavailable for insertion, the resulting k_{eff} for 19 rods in the core would be 1.00.

The additional reactivity due to the presence of steam in the core is relatively small. The reactivity addition is essentially linear with steam pressure at approximately 0.05% per 100 psi. However, the addition of steam to the reactor would raise coolant system pressure and would be accompanied by a reactor coolant high pressure scram. This scram would occur when the partial pressure of the steam in the core is approximately 50 psi (with system initially at 20%

power). Thus, the maximum amount of reactivity that could be added to the operating reactor due to steam addition is approximately 0.025%. This reactivity addition is less than could occur due to improper fuel handling and consequently is not considered further.

8.6.7 Withdrawal of One Control Rod By Service Machine

The service machine will be routinely used for service operations in the reactor vessel top plenum while the reactor is operating at power. For any service operation requiring the use of the service machine fuel chute, a control rod shield plug which includes the control rod drive plug and the associated control rod and shroud must be removed. The normal procedure will be for the reactor operator to withdraw the control rod to the upper limit. The control rod is then further withdrawn using the control rod drive manual drive and is finally locked to the shield plug. The service machine is then attached to the nozzle and the plug and control rod are withdrawn.

Using this procedure, the reactor operator will be in direct control of that portion of the operation which changes core reactivity; i.e., the "withdraw" to the normal upper limit. However, this procedure is not followed in the event of a control rod cable failure. Operating procedure requires that the reactor be shut down to retrieve a control rod which has fallen into the core. Although removal of a dropped rod while the reactor is operating is a direct violation of operating procedures, the withdrawal of one control rod from the bottom of the core at an assumed rate of 30 in./min was simulated at 20% and 100% reactor power with both manual and automatic blower control and with flux minus flow and high outlet coolant temperature scrams. This rate is 50% greater than the rate at which the service machine can withdraw a control rod.

Figures 8.6.7.1 and 8.6.7.2 show the results of the simulation for 20% power operation. A maximum fuel cladding temperature of 1215 F occurs in the average power channel. If the automatic flow control is selected at the time the reactivity insertion occurs, the maximum cladding temperatures are correspondingly lower.

Figures 8.6.7.3 and 8.6.7.4 show the results of the simulation for 100% power operation. The maximum fuel cladding temperature in the average power channel is 1438 F. No fuel element failures will occur and consequently, withdrawal of a control rod by the service machine does not represent a hazard to either the public or to the operating personnel.

8.7 Electrical System Failures

8.7.1 General

The plant electrical system is so designed that no single failure can cause a complete plant power outage, except a loss of the 13.8 kv plant bus itself, the loss of the external network power while the station generator is not operating, or the loss of the station generator while the plant is isolated from the external network. Most single failures cause a partial outage and a substantial part of the plant equipment may become inoperative. Electrical distribution within the plant is so arranged that all vital equipment is duplicated and is supplied power from separate buses; in addition, this vital equipment is protected from potential damage from missiles, fire, flooding, excessive temperatures and moisture. The tie breaker, normally closed, between the separate buses of the 2.4 kv switchgear must be manually closed in the event it has been tripped open due to a malfunction or fault (other than a bus fault) which has deenergized a bus section. Relay action is discussed in Section 5.7.4.

The 480 v load centers are powered from station service transformers during

be removed.

8.9.4 Closure of Coolant Circuit Isolation Valves

Both EGCR coolant circuits are equipped with isolation valves in the reactor vessel inlet and outlet lines. These valves can be used to isolate one coolant circuit from the remainder of the reactor coolant system. These valves can be remotely and manually controlled by the operator and are also closed automatically by the steam generator isolation system in event of a steam generator malfunction. Interlocks insure that the valves in a coolant circuit cannot be closed while the reactor coolant blower in that circuit is operating. However, there is always a possibility that flow in one circuit can be reduced due to a valve or control system failure.

From the blower characteristic curve (Figure 5.4.5.1), it can be seen that a 15% reduction of the operating flow through the blowers causes the pressure rise across the blowers to increase to the postulated surge line when operating at 100% speed.⁽³⁴⁾ (This percentage increases with decreasing blower speed.) With one valve closing, the postulated surge line is reached in 50 sec; while with two valves closing, this point is reached in 42 sec (Figure 8.9.4.1). The blower in the normal circuit would continue to operate normally, but with an increase in flow and a corresponding decrease in pressure rise as the flow through the blower in the abnormal loop decreases. The flow in the normal circuit has increased approximately 5% just prior to the blower in the abnormal circuit reaching the postulated surge line. Shop tests on the first blower unit⁽⁴⁷⁾ show that the expected surge line is above the postulated surge line shown in Figure 5.4.5.1. Therefore, the results using the postulated surge line are in the safe direction.

Just prior to reaching the postulated surge line, the total flow is:

$$W_T = \frac{W_1}{W_{n1}} + \frac{W_2}{W_{n2}} = \frac{0.85 + 1.05}{2} = 0.95 \quad \text{Eq. (8.9.4)}$$

where:

- W_1 is the flow through the abnormal circuit
- W_2 is the flow in the normal circuit
- W_n is the rated circuit flow at 100% power
- W_T is the total coolant flow.

The flux-minus-flow comparator generates a scram signal only when the difference equals or exceeds the reactor power value by 10%. When the postulated surge line is reached it is assumed that the flow from the blower in the abnormal circuit is zero. When this occurs the total flow undergoes a step change to 55.2% of full flow and the reactor is scrammed.

The results of the transient temperature analyses are shown in Figure 8.9.4.2 for each of ten axial positions along the maximum power fuel channel. The maximum nominal temperature occurs at position 5, which is in the third fuel assembly from the bottom of the core and occurs approximately one second after the accident. The maximum nominal fuel cladding temperature reached is 1575 F. The maximum temperature of the remaining positions is below this value.

As discussed in Section 8.3.1, the long term fuel failure criterion is 1800 F. Since coolant system pressure is not lost and the temperature transient is of short duration, no fuel element failures are expected.

Figure 8.9.4.3 shows the reactor coolant inlet and outlet temperatures. The maximum outlet temperature of 1106 F is reached five seconds after the accident. The reactor inlet temperature drops steadily from the beginning of the accident.

EFFECT OF ISOLATION VALVES UPON FLOW IN THE ABNORMAL LOOP

FIG. 8.9.4.1
8-46

FIG. 8.9.4.2
8-47

REACTOR COOLANT TEMPERATURES FOLLOWING ACCIDENT

FIG. 8.9.4.3.

The transient temperature analysis assumed that the reactor was operating initially at 100% power but with 92.5% of design flow. This corresponds to a condition where flow in one coolant circuit is 100% of full flow but the flow in the other circuit is 85% of full flow. This neglects the 5% flow increase through the normal circuit due to decreased total system flow and assumes that the blower in the affected loop is at the surge point. The scram is assumed to occur 0.5 sec after the blower in the abnormal loop surges. The accident would actually start with both circuit flows approximately equal. As the valve or valves continue to close, the total flow would decrease with a corresponding increase in core temperatures. The analysis, however, assumed that the reactor had been previously operating on a steady-state basis with the reduced flow (92.5% total) and consequently with higher initial temperatures than would actually exist. This assumption results in conservative conclusions.

It is concluded that no serious hazard to operating personnel or the general public exists in the highly improbable event of the isolation of one coolant circuit while the reactor is operating at 100% power.

8.9.5 Single Blower Failure

Each of the two reactor coolant circuits includes a 10 stage axial flow blower. The control and operation of each blower is through independent controllers, with only the average reactor coolant outlet temperature signal common to both controllers. A component failure within either the drive motor or control equipment, the hydraulic coupling, the speed increaser, the blower itself, or the blower speed controller could cause that blower to stop. Fatigue failure of a single blower rotor blade or interference between rotor and stator could cause the stripping of a number of blades. The fatigue failure of the blower rotor or stator blading is unlikely. The blading material, the blade design, and the inspection quality have been so executed that the cyclic stress levels to be expected during blower excursions in the normal operating region are much less than the endurance limit of the blades.⁽⁴⁸⁾ It is therefore expected that the blower will withstand the stresses of surging for an indefinite period of time without damage.

An analysis was performed of the temperature transients resulting from a failure of a single blower.⁽³⁵⁾ To derive a conservative result which would represent the most severe possible failure involving one blower, it was assumed that, following the instantaneous failure of a single blower, the failed blower offered no resistance to fluid flow. As result of these assumptions, the coolant flow through the reactor is instantly reduced to 29% of the initial flow. Figure 8.9.5.1 shows the maximum fuel cladding temperatures after the accident for the maximum power fuel channel and for the average power fuel channel. The maximum fuel element temperatures do not exceed the long-time fuel element failure criterion of 1800 F, and therefore no fuel failures are expected as the result of this accident. Figure 8.9.5.2 shows the reactor outlet and fuel channel outlet temperatures following the accident. The maximum outlet temperature does not exceed the hot leg piping long term failure criterion of 1150 F.

8.9.6 Power Failure

A power failure on the 13.8 kv plant bus results in interruption of power to all plant equipment which is not powered by the failure-free systems. The reactor is scrammed by the loss of power. Within approximately 27 sec, the diesel generators are at rated frequency and voltage to supply 480 v power to the emergency load center. Once emergency power is available, the vessel nozzles and service machine are supplied with helium circulated by the vessel cooling system as described in Sections 5.4.3 and 5.7.5.

The need for forced cooling during a plant power failure is dictated by the

heating of the reactor vessel top nozzles. The reactor core can be adequately cooled by natural circulation of helium through the reactor coolant system following a blower coastdown, since normal operating pressure is maintained. The heating rate of the top nozzles without forced cooling of the reactor has

FIG. 8.9.5.1
8-50

significant heatup of the reactor core will occur for one hour without core cooling.⁽⁴²⁾

Isolation of a reactor coolant loop can be reversed by either returning the manual isolate switch to the "off" position or by turning the "override automatic isolation" switch to the "override" position, depending on what caused the isolation. This will enable the operator to reopen the valves by manipulating the individual control switches (eight per loop) and restart the reactor coolant blower.

8.10 Loss of Coolant Accidents

8.10.1 General

A loss of coolant, i.e., loss of pressure, accident occurs as a result of any accidental or inadvertent opening or breach of the reactor coolant system. The significant variables of such an accident are the size and location of the opening. Small openings in the reactor coolant system cause the system pressure to decrease slowly. Reactor coolant system mass flow decreases at the same rate as the coolant system pressure. A low pressure scram occurs when the coolant system pressure decreases to 225 psig. Larger openings in the reactor coolant system cause the coolant pressure to decrease rapidly with a resulting decrease in core cooling. Rapid system depressurizations may also produce large pressure differentials across reactor vessel and coolant system components and temporarily interfere with the normal flow.

Loss of coolant accidents have been the subject of extensive analyses throughout the design of the EGCR. Because of the seriousness of these accidents, strong emphasis has been placed upon strict fabrication and inspection procedures for all pressure containing reactor coolant system components, particularly the reactor pressure vessel.

Extreme care has been exercised in the design of the reactor pressure vessel. The design has been thoroughly analyzed through rigorous mathematical analysis and further confirmed through research and development with scale model analysis. The analysis covers both the steady-state pressure and thermal stresses and transient thermal stresses. Extensive strain gages analysis on the actual vessel confirm the design analysis (Section 4.5).

A stringent quality control program is being followed throughout the fabrication and erection of the pressure vessel, its internals and appurtenances to insure that the completed vessel will be of superior quality (Section 4.5.3).

Failure of the reactor pressure vessel shell is not considered credible, since:

- a. Pressure vessel technology has been sufficiently established and demonstrated by past experience, and the vessel is constructed in accordance with a recognized code which requires large safety factors.
- b. The stresses have been thoroughly analyzed, both mathematically and experimentally.
- c. The loads which form the input to the analysis are well known.
- d. Tests at greater than design pressure, ultrasonic examination of materials, and magnetic particle, ultrasonic and radiographic examination of welds prove the integrity of the vessel.

The design, fabrication, quality control and inspection of the reactor coolant system piping and the other components therein are such as to minimize the

number and size of defects which may be introduced into the system. Thus, every effort is made to negate failures by the elimination of detectable defects or deficiencies which, if not eliminated, could lead to failures in the reactor coolant system piping. Therefore, the probability of piping failures is extremely small.

From a failure probability point of view, it is considered that a circumferential crack may possibly occur at one of the following locations in the reactor coolant system:

- a. Dissimilar metal welds in pipe
- b. Attachment welds of piping to valves or other components that are subjected to high transient thermal stresses
- c. Expansion joints.

Should the crack propagate completely around the pipe, it would result in a separation of the pipe ends at the failure location. Motion limiters are employed around the hot leg piping such that the severed pipe ends cannot be displaced more than 1 in. beyond the limits of thermal movement. Motion limiters are also employed on the cold leg piping from the reactor vessel to the coolant inlet isolation valve. The cold leg piping between the inlet isolation valve and the steam generator is not equipped with motion limiters.

The presence of defects in the reactor coolant system piping could possibly result in failures at locations other than those considered for circumferential cracks. A failure which originates from any defect would appear first as a longitudinal crack which may eventually assume an elliptical shape or opening. The maximum size of the opening can be calculated based upon the strain energy present in a pressurized system. The detailed analysis of the piping system derives the stresses which form the input to the determination of the maximum credible size of ductile tear at various locations in the piping system. The credible size openings are discussed in Section 8.10.2.

8.10.2 Reactor Coolant System Piping Failures

Both the size and location of a piping failure are significant when determining the effect of such failures upon the reactor coolant system. To establish a basis for determining the structural adequacy of reactor coolant system components, it is first necessary to establish the location and largest credible size of the failure.

The reactor coolant system piping was examined to determine the largest credible system breach for each type of failure, i.e., circumferential severance and ductile (longitudinal) tear. Table 8.10.2 summarizes the results obtained.

Detailed analyses were performed to determine the pressure differentials developed across system components in the event that the credible circumferential type breaks were to occur in either a hot or cold leg of the reactor coolant system.

It is not necessary to analyze each failure in the piping system to establish the maximum pressure differentials that could be developed across core and reactor pressure vessel components, since for any given size breach the further the breach is from the reactor vessel the lower will be the pressure differentials developed across these components.

TABLE 8.10.2

Coolant System Piping Failures Coolant Escape Areas^a

<u>Location</u>	<u>Largest Credible Opening</u>	
	<u>Circumferential (in.²)</u>	<u>Ductile (Longitudinal) (in.²)</u>
<u>Hot Leg</u>		
Dissimilar metal welds in 31-in. hot leg pipe	75	--
Connections to hot leg isolation valves	299	--
Attemperator outlet	314	--
Piping (maximum value for 36-in., 31-in., 24-in. O.D. pipe)	--	24
<u>Cold Leg</u>		
Inlet end of isolation valves (total area of both pipe ends)	710	--
Piping (22 in. O.D.)	--	9

^aCredible breaks at all locations have not been included in the above table since they have been considered and do not result in flows that give the maximum differential pressure cases for the core and reactor pressure vessel components.

The following assumptions were made in the analysis of these accidents:

- The rupture occurs instantaneously.
- The reactor coolant undergoes adiabatic expansions and compressions.
- No flow occurs through the reactor coolant blowers during the depressurization.
- No effect from a steam generator leak in a failed steam generator is considered since failure will be sequential to the depressurization.
- The coolant system is idealized as several gas volumes with fluid flow resistances connecting the volumes.
- Coolant escapes from the piping failure at sonic velocity.

The assumption that there is no flow through the reactor coolant blowers during the accident results in an overestimate of the pressure differentials developed across reactor core and vessel components, since the reactor coolant can then leave the reactor vessel by only one path. This assumption, however, permits comparison of depressurization accidents to the accident for which the detailed analysis was performed by comparing only the initial exhaust mass flow leaving the coolant system. With the above assumptions, for a failure in a given portion of the coolant system, flow can be only in one direction, that is toward the rupture. Therefore, the initial exhaust flow from the rupture indicates the magnitude of the pressure differentials associated with a depressurization due to a particular size failure.

Figures 8.10.2.1 and 8.10.2.2 show the pressure differential developed across the reactor core and the exhaust flow for the largest circumferential failure credible at the attemperator outlet connection weld in the hot leg. The maximum pressure differential developed across the reactor core is 24.3 psi upward; the maximum exhaust flow is 550 lb/sec. The time to depressurize the coolant system is estimated to be approximately 10 sec.

CORE PRESSURE DIFFERENTIAL
FOLLOWING HOT LEG RUPTURE
JUST AFTER ATTEMPERATOR
(314 IN² OPENING)

FIG. 8.10.2.1

EXHAUST FLOW FOLLOWING
RUPTURE AFTER ATTEMPERATOR
(314 IN² OPENING)

FIG. 8.10.2.2

The top plenum of the reactor vessel depressurizes faster than the bottom plenum if the rupture is in the hot leg since it is closer to the rupture; consequently, the pressure differentials developed produce an upward force on core components. To prevent levitation during such an accident, the bottom core plate is bolted to the core support grid. The support grid is welded to the top of the support skirt which is welded to the bottom head pressure vessel.

The graphite core structure is not rigidly fastened to the core bottom support plate but rests upon the plate. Lifting of the core graphite during a depressurization, however, does not occur, since the area available for the pressure differential to act upon is limited by the gas seals at the bottom of each fuel channel (Section 4.3.2). A pressure differential across the graphite columns and core seals in excess of 100 psi is required to levitate the columns. No credible accident results in such high pressure differential.

The column of fuel is supported within the channel by a bottom dummy assembly which is locked to the core bottom plate and by a top dummy assembly which is locked to the core fuel channel extension which is attached to the top grid structure intermediate plate. The calculated pressure differential required to overcome the weight of the fuel assemblies and top dummies, top grid structure and appendages is in excess of 29.5 psi, neglecting friction and structural rigidity.

Although a serious depressurization accident is extremely unlikely, there is a possibility that the charge machine could be attached to a fuel channel when a depressurization accident occurs. To prevent fuel assemblies from leaving the fuel channel during a depressurization, the top dummy is designed so that an upward force on the top dummy while the latch release is in the locked position applies a large locking force, precluding unlatching of the top dummy. Pressure differentials as large as 300 psi cannot unlatch the top dummy assembly. Although high pressure differentials can be developed across the one fuel channel in the reactor, lifting of the graphite column does not occur.⁽⁴³⁾

Figures 8.10.2.3 and 8.10.2.4 show the pressure differential across the reactor core and the flow leaving the reactor side of the rupture respectively for a circumferential failure upstream of the inlet isolation valve.⁽⁴⁴⁾ The maximum pressure differential developed across the reactor core is 59.7 psi; the initial (maximum) flow leaving the reactor is 661 lb/sec. The depressurization time for this accident is estimated to be 3 to 5 sec. This failure results in depressurization of the bottom plenum of the reactor vessel faster than the top plenum so that the pressure differential acts downward upon the core structure.

The maximum pressure differential developed across the reactor temperature barrier is approximately equal to the pressure differential across the reactor core.⁽³⁷⁾ The maximum differential pressure developed across the reactor coolant blower in the sound loop is associated with a rupture of the reactor inlet piping and is less than 61.2 psi. The blower is designed to withstand this pressure differential. The maximum differential across the steam generator in the sound loop is less than 2 psi and occurs as a result of a circumferential failure in the hot leg piping.

There is no credible piping failure which can result in pressure differentials across the core or the reactor coolant system components in the sound loop that exceed those associated with the largest credible circumferential rupture. Reactor vessel and core components are designed to withstand the maximum pressure differentials resulting from all credible ruptures in either loop. The reactor coolant blowers and steam generators are designed to withstand pressure differentials resulting from all credible ruptures in the opposite loop.

8.10.3 Reactor Coolant System Relief Valve Failure

Three relief valves protect the reactor coolant system from over-pressure and are located on the cold leg piping. One relief valve is located at the coolant gas outlet of each steam generator and the third is located on one reactor inlet pipe on the reactor side of the reactor coolant circuit isolation valve. Relieving pressures are 330 and 340 psig for the two valves located just downstream of the steam generators and 350 psig (reactor coolant system design pressure) for the valve located on the reactor pressure vessel inlet. Each valve is sized to relieve the maximum steam flow vented into the reactor coolant system as a consequence of a steam generator internal failure (Section 8.11.7). The valves discharge through a common header leading to the vent system. A parallel set of valves is installed for stand-by service; however, only one set is available at any time.

Valve failure can be of two modes and of varying degrees within each mode, i.e., a valve may fail to open or to close, or it may partially open or close. Failure to open in the event of an over-pressure does not constitute a hazard since one of the remaining relief valves opens at a slightly higher pressure but within the design limits of the coolant system. Further, since each valve is capable of handling the maximum relief flow requirement no over-pressure can result.

If a valve failure occurs as a result of a spring failure, a partial depressurization occurs. This depressurization could result in a small core coolant flow disturbance. No fuel element failures should occur since adequate core cooling exists. The stand-by set of relief valves is connected into the system when the coolant system has depressurized to 270 psig. The first set is simultaneously valved off.

Similarly, failure of a valve to reseal tightly does not create a hazard to the public. The unseated relief area is small and the available capacity of the coolant makeup system is adequate to maintain coolant system pressure. Radioactive helium leaks from the reactor coolant system; however, if the gaseous activity concentration in the stack is in excess of the allowable, the containment shell is isolated automatically.

8.10.4 Auxiliary Systems Failure

A reactor system depressurization accident can occur as the result of piping failures in auxiliary systems. The severity of the consequences of such failures is limited by the smaller piping sizes used for these systems.

The most serious failure of an auxiliary system is a major failure of either the charge or service machine attachment to the reactor vessel.

The limiting opening in the reactor coolant system as a consequence of attachment failure is the cross sectional flow area of a top or bottom reactor vessel nozzle. Analysis of initial flows⁽⁴³⁾ from these openings indicates that the differential pressures produced across the system components do not exceed design values. System depressurization time is estimated to be no less than 25 sec.

8.10.5 Consequences of Coolant System Failures

Failure of the reactor outlet piping near the reactor vessel produces an upward pressure differential across the core, while failure of the inlet piping near the reactor vessel produces a downward pressure differential. Between these extremes there are combinations of failure size and location which could produce a condition such that little or no pressure differential would exist

across the reactor core. Such a condition could persist for a period of up to 30 sec until the system is depressurized. Thus the most severe accident from the point of view of fuel element failures is a coolant system depressurization such that coolant flow through the core is suppressed during the depressurization. A breach of approximately 60 in.² in the reactor inlet coolant piping near the blower discharge would have this effect. The initial discharge from this size opening is approximately 120 lb/sec, which is the normal reactor coolant flow at 100% design power. This type of depressurization corresponds to an instantaneous loss-of-flow accident for the short period of time.

On the other hand, it has been assumed that large piping failures might be accompanied by internal damage to the steam generator nearest the failure. This internal damage may result from the large pressure differentials across the tube bank, baffles, or shroud. Such a failure would allow the contents of the steam generator to blow down into the containment shell. This release of steam, along with the coolant system helium inventory, causes a pressure increase in the containment shell.

A rupture of the reactor coolant system could allow the admission of air from the containment shell into the reactor coolant system. Any introduction of air will cause some degree of oxidation of the core graphite with the attendant release of chemical heat. This heat release due to graphite oxidation is considered in the analysis of depressurization accidents.

The analysis of the most severe depressurization accident was divided into three major parts. The first part of the analysis determined the fuel element cladding temperatures for the first 30 sec following the depressurization. This represents the most severe core cooling condition possible, i.e., no flow.

The second part determined the behavior of core temperatures for various possible conditions for the period of time from 30 sec to one hour following the accident. It was assumed that one hour is sufficient time for the reactor operator to determine the nature of the accident and decide upon and execute remedial action.

The third part described core temperatures from one hour after the accident.

Figure 8.10.5.1 shows the temperatures of the fuel element cladding for the first 30 sec of the accident. During this period, core flow is assumed to be zero. Approximately 330 fuel element failures occur.⁽¹⁾ A scram delay of 5 sec was used. This has been estimated to be the longest scram delay for a depressurization.

From 30 sec to one hour, temperature in the core will vary depending upon the amount of coolant flow through the core. Following any credible depressurization accident the reactor coolant blower in the sound loop will be operating at the ramp down speed. Following many of the depressurization accidents both of the blowers would be operating. The core cooling available during the first hour is not only dependent upon how many blowers are operating but upon the gas being circulated, the position of isolation valves in the failed loop and the operation of the vessel cooling compressors. Graphite temperature in the core during the first hour was determined by assuming a number of conditions of equipment operation to determine the effect of various core flows on graphite temperature. All the cases postulated represented sequences of events that most likely could not occur at the times assumed in any credible accident, but were used in order to have a very conservative accident analysis. Figure 8.10.5.2 shows the fuel assembly graphite sleeve temperatures for the various cooling conditions indicated below:⁽¹³⁾

<u>Case</u>	<u>Core Coolant Flow (lb/hr)</u>	<u>Case</u>	<u>Core Coolant Flow (lb/hr)</u>
No Cooling	0		
1 B	8014	8 B	10,714
2 B	5298	12 B	3410
3 B	4590	15 B	2000
4 B	6860	25 B	1000
5 B	3948		

The sleeve temperature transients for the above cases all fall within the shaded area shown on Figure 8.10.5.2. Each case represents a distinct arrangement of valves, location of failure and operating blower or vessel cooling compressor. Temperatures are generally higher for flows in the neighborhood of 2000 to 3000 lb/hr. As a further degree of conservatism, an arbitrary 50 F was added to the highest temperatures (Figure 8.10.5.2). The curve through these points was then used as input data to the next part of the calculation. In order to have a thorough understanding of the effects upon the core and the graphite structure subsequent to the depressurization accidents, it was necessary to evaluate the conditions through a very extensive analysis using the various modes or methods of core cooling that would be available subsequent to the accidents. In addition, it was necessary to consider various flows and the use of various combinations of the equipment to insure that adequate cooling is available for after-heat removal at such rates of cooling that chemical reaction may be controlled within safe limits for all credible accidents.

The analyses show that graphite oxidation in the EGCR following a depressurization accident can be controlled under the following conditions: (13)(45)

- Immediately following a depressurization accident, graphite oxidation can be controlled with a core coolant air flow in excess of 30,000 lb/hr.
- Thirty minutes after a coolant system depressurization and assuming the most severe conditions existed during the thirty minutes following the depressurization, the core coolant air flow would have to be greater than 55,000 lb/hr to prevent runaway graphite oxidation.
- Sixty minutes after a coolant system depressurization and assuming the most severe conditions existed during the sixty minutes following the depressurization, the core coolant flow would have to be greater than 60,000 lb/hr of air or 8000 lb/hr of nitrogen (less than 1 wt % air) to prevent runaway graphite oxidation.

Adequate core cooling in accordance with the requirements indicated above is provided by one of the following methods:

- Continuous operation of both of the reactor coolant blowers at the ramp-down speed
- Operation of either of the reactor coolant blowers at an increased speed
- Operation of one of the two compressors in the emergency cooling system. Nitrogen purge is required if this system is used prior to three days after the accident.

The actual method used to cool the core will depend upon the conditions following any depressurization accident. The procedures followed by the operator during such a postulated accident are discussed in Section 16.10.

After graphite temperature in the core has been reduced using the required flows indicated previously, the flow requirements are dictated primarily by the requirements to remove decay heat. After three days, the flow requirement is reduced to approximately 8,000 lb/hr of air. Such a flow can be provided by the emergency cooling system (with no nitrogen purge), the vessel cooling compressor, or either of the reactor coolant blowers at reduced speed.

Figure 8.10.5.3 shows the fuel element cladding temperature with a coolant flow of 60,000 lb/hr of air from one hour after the accident. Figure 8.10.5.4 shows the fuel element cladding temperature with a coolant flow of 8000 lb/hr of a

Depressurization Accident; Fuel Clad Temperature Transient.

FIG. 8.10.5.3

8000 lb/hr OF NITROGEN-AIR MIXTURE
ONE HOUR AFTER ACCIDENT

Depressurization Accident ; Fuel Clad Temperature Transient .

FIG. 8.10.5.4

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nitrogen-air mixture from one hour after the accident. Figure 8.10.5.5 shows fuel element cladding temperature with a coolant flow of 8000 lb/hr of air from 3 days after the accident.

As previously mentioned, it has been assumed for purposes of analysis that a depressurization accident may be accompanied by a steam generator internal failure which will allow some steam or water to enter the reactor core. Consequently, reaction between steam and graphite may occur producing some hydrogen. When one or both reactor coolant blowers are operating to provide adequate core flow to prevent runaway graphite oxidation, hydrogen generation will be limited so that combustion is not possible.

If the emergency cooling system is used to cool the core, hydrogen generation is suppressed by the nitrogen purge. Any hydrogen formed would be diluted by the nitrogen to prevent a combustible mixture.

8.10.6 Cooling Capability Following Depressurization Accidents

As discussed in Section 8.10.5 the initial number of fuel element failures associated with a loss of coolant accident depends upon the size and location of the coolant system rupture. If it is postulated that forced cooling to the core is completely lost during the depressurization accident, it can be shown that this flow can be discontinued for a period of one hour without the rupture of any additional fuel elements. However, if forced cooling is not reestablished, the fuel element cladding would eventually reach excessively high temperatures due to decay heat and gross failure of fuel elements would occur.

The design of the EGCR is such that there are four systems normally available to supply effective core cooling flow following reactor shutdown for any credible accident condition. These systems are:

- a. Two identical reactor coolant circuit systems
- b. The vessel cooling system with an operating compressor and a standby compressor
- c. The emergency cooling loop system with two standby compressors.

Depending upon the size and location of the coolant system rupture, all four systems may remain operable; it is highly probable that three systems will be operable; however, at least two will be available.

If the depressurization is slow, pressure differentials across the steam generator and blower in the failed loop will not be large and consequently both reactor coolant circuits and the vessel cooling system are available in addition to the emergency coolant loop system. On the other hand, it has been postulated that rapid depressurizations may cause internal damage to the steam generator or to the blower in the circuit containing the rupture. The coolant circuit containing the failure, therefore, is assumed to be of no value as far as heat removal capability is concerned, even though the blower in that circuit may continue to operate and remove heat from the core.

If the rupture is in the cold leg the blower in the sound coolant circuit will not be damaged. The vessel cooling compressors will not be damaged. However, since the possibility of tripping the vessel cooling system excess flow valves exists, neither vessel cooling compressor may be available for forced core cooling until the reactor operator re-positions the excess flow valves and restarts one vessel cooling compressor. The emergency cooling loop system will not be affected by the depressurization since it is isolated from the reactor coolant system during reactor operation.

Thus, following a rapid depressurization as the result of a cold leg rupture there are available:

Depressurization Accident; Fuel Clad Temperature Transient.

FIG. 8.10.5.5

- a. One reactor coolant blower operating at the ramp down speed.
- b. The emergency cooling loop system available for operation.
- c. The vessel cooling system available for operation or operating.

A rapid depressurization as the result of a hot leg rupture will tend to force the reactor coolant blowers into surge. Even if the blower surges during the depressurization of the coolant system, blading damage would not occur and the blower would be operable. The compressors in the vessel cooling system will not be damaged; however, the excess flow valves in the vessel cooling system may close. This will require the operator to re-position the excess flow valves and re-start one compressor before forced cooling using the vessel cooling system can be established. Again, the emergency cooling loop is isolated during operation of the reactor and cannot be damaged by the depressurization. Thus, following a rapid depressurization as the result of a hot leg rupture there are available:

- a. One reactor coolant blower operating at the ramp down speed.
- b. The emergency cooling loop system available for operation.
- c. The vessel cooling system available for operation.

Thus following any depressurization accident, one reactor coolant blower will be operating, the emergency cooling loop system will always be available, and the vessel cooling system will be available. Postulating the most severe core cooling sequence, the vessel cooling system cannot be used to cool the reactor core until three days following the accident (Section 8.10.5). The emergency cooling system contains two compressors, thus, for the first three days there will always be at least two compressors to provide backup forced core cooling.

8.11 Accidents Involving Steam System and Plant Auxiliaries

8.11.1 Minor Accidents

Several minor failures can occur in the steam plant and plant auxiliary equipment. None of these failures results in a radiological hazard. Certain failures require the reactor 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.

a. Steam Tube Rupture in Low-Pressure Heater

The low-pressure heater is supplied with steam from the turbine 16th stage extraction. Extraction pressure varies with turbine load and is 14.0 psia at 100% turbine power. Condensate enters the low pressure heater at a pressure greater than 60 psia. In the event of a piping rupture, water flows to the steam side of the heater, causing the drain water level in the heater to rise. Drain water flow back to the main condenser is controlled by a level control valve which receives its signal from a level control in the heater. If the rupture is of such a magnitude that the level controller cannot maintain the required level, an alarm is actuated. A greater than normal return flow to the main condenser causes the deaerator level to drop. If the level drops to a pre-set low level the main feedwater pump is tripped. Trip of the main feedwater pump results in a low steam drum level which initiates a reactor scram and start-up of the shutdown feedwater pumps. No radiological hazards occur as a result of this failure.

b. Loss of Condenser Vacuum

The air ejector removes non-condensables from the main condenser. Any component failure which interferes with this function tends to reduce condenser vacuum and may eventually result in low vacuum trip of the turbine. The loss of condenser vacuum also results in closure of the

<u>Equipment or Space</u>	<u>Most Likely Types of Fire</u>	<u>Protection</u>
Yard Buildings		
Chlorination building	Miscellaneous	Portable 15-lb CO ₂ extinguisher
Guardhouse	Paper	Portable 15-lb CO ₂ extinguisher
Yard	Vehicular fires, outdoor storage areas, building roofs	Fire hydrants, hoses, and combination nozzles

8.13.3 Protection Against Sabotage

The following measures will be employed to minimize the possibility of sabotage to the EGCR. The security arrangement which would apply during a national emergency is classified and is not discussed here.

a. Access Control - Measures to prevent or detect unauthorized entry.

1. Perimeter fencing encloses the entire facility, with a single point of entry at the main entrance gate.
2. Vehicle parking area is outside the perimeter fence adjacent to the entrance gate.
3. A minimum of two guards will be posted during normal working hours (8 hr/day, 7 days/week), one located at the main entrance gate and the other acting as patrol or escort. On other shifts the entrance gate is locked and access is controlled by the operations supervisor on shift.
4. Employees and visitors will be required to wear identification badges.
5. All plant employees will be instructed, and periodically reminded, to be alert for and to report the presence of any person who is not clearly identified and authorized.
6. The shift supervisor on duty will control access into the reactor building, the control room and the exposed cable area under the control room.
7. The AEC patrol covers the roads in the adjacent area to restrict public access to certain designated routes through the controlled area. The AEC controlled area includes all land surrounding the EGCR site but not the Clinch River (Melton Hill Lake) which is the AEC boundary. The domestic water line and the 13.8 kv transmission line are wholly within the AEC controlled area.

b. Control of Visitors and Non-Employees - Measures to limit access to authorized areas only.

1. A register at the main entrance gate will record the entry of all visitors and non-employees.
2. Visitors and non-employees will be admitted only by authorization of the project manager or his representative.
3. Visitors and non-employees will be provided with an escort, as required, while they are in the plant area. In general, surveillance will be required on a continuous basis in all plant areas, except the office area.
4. Contractor employees or other non-TVA employees, who are on official business at the project will not be escorted.

- c. Personnel Security - Measures to minimize the possibility of sabotage by employees.
 - 1. Each local employee will have had a national agency check with written inquiries (equivalent to AEC "L" clearance).
 - 2. At least two employees will be on duty in the main control room at all times.
- d. Design - Features which are also protective in nature.
 - 1. The reactor safety system acts to minimize the damage resulting from any act of sabotage.
 - 2. The plant annunciation system warns the operators of equipment malfunction.
 - 3. Any opening of the access doors to the reactor building is annunciated in the main control room.

From the above it may be noted that adequate personnel and visitor control policies are being established to make severe damage to the facility by sabotage from within extremely unlikely. Since the site is within a well established AEC controlled area the established procedures are also adequate to make severe damage to the facility by sabotage from external sources extremely unlikely.

8.13.4 Severe Weather

The EGCR structures are designed to withstand the wind and snow loads which occur at Oak Ridge. Lightning which strikes the containment shell is conducted to the ground through the steel shell without damage to the shell or to the interior of the building. The elevated water storage tank and the exhaust stack are protected by lightning rods on top of the structures.

A storm drainage system carries rainfall runoff beyond the plant limits. The plant is located well above the highest possible flood level of the Melton Hill Lake. The yard piping and electrical systems and the electrical substation are adequately designed to avoid damage from freezing.

8.14 Identification of the Maximum Credible Accident

The maximum credible accident is defined as that accident whose consequences in terms of dose to the public are not exceeded by any other known credible accident. The most severe class of accidents that can occur in the EGCR is the loss of coolant (depressurization) accidents. These accidents, in general, result in the most severe fuel element temperature transients. In addition to these temperature transients, the external pressure on the fuel element cladding is reduced thereby changing the force acting on the cladding from an external or collapsing force to an internal or bursting force. This internal pressure combined with the severe temperature transients results in the largest number of predicted fuel element failures.

The maximum credible accident, however, cannot be determined solely on the basis of fuel element failures since these govern only the release of activity to the containment shell. It is quite possible that the accident which results in the largest activity release from the reactor may not be the accident which results in the greatest release of activity beyond the containment shell. For this reason, the maximum credible accident in the EGCR is a composite accident, and includes the most severe features of several depressurization accidents.

The MCA for the EGCR is identified as a depressurization accident which suppresses core coolant flow for 30 sec and causes internal failure of the steam generator in that coolant circuit which contains the failure. The first feature would be associated with only a relatively small rupture, located between the blower and the reactor vessel, while the second, if possible at all, could be caused by only a large rupture adjacent to the steam generator. From the end of the depressurization to the end of the accident the most severe reactor condition from the standpoint of reactor cooling is assumed (Section 8.10.5). From one hour to several days after the accident, the emergency cooling loop and the nitrogen purge gas system to control graphite oxidation are assumed to be in operation.

Approximately 330 fuel elements fail during the first 30 sec of the accident. No subsequent fuel element failures occur.⁽¹⁾ Concurrent with controlled graphite oxidation, the UO₂ within the failed fuel elements is assumed to oxidize at the rate given in Section 8.4.1.

The containment shell internal pressure transient following the MCA is given in Section 8.15. However, for purposes of calculating activity release from the containment shell, a constant internal pressure of 9 psig is assumed for the first 24 hr after the accident. It is next assumed that at 24 hr after the accident the containment shell internal pressure is reduced to 4.2 psig and remains constant from 24 to 72 hr after the accident. Between 72 and 81 hr after the accident, the pressure in the reactor building is assumed to be 0.75 psig. At 81 hr after the accident the pressure in the containment shell is assumed to be reduced to zero psig. Containment shell leakage is based on 0.3% of the contained mixture per day at 9 psig.

The assumed containment shell internal pressures are conservative, based upon the calculated pressure transient following the accident, and the capacity of the containment shell fission product removal system. It is assumed that the reactor operator places the fission product removal system into operation 24 hr after the accident.

For the first 24 hr after the accident, the dose to the public is due only to the leakage (assumed to be at ground level) from the containment shell. From 24 to 81 hr, the dose is due both to containment shell leakage and the effluent from the fission product removal system (at stack level). The stack discharge rate is conservatively assumed to be 512 SCFM from 24 to 72 hrs, and 217 SCFM from 72 to 81 hrs after the accident.

After 3 days the nitrogen purge is stopped; consequently, reactor building pressure decreases to atmospheric at about 81 hours. The effluent from the stack and the activity release from the containment shell stop at 81 hrs.

The requirement for afterheat removal from the reactor core terminates at approximately 3 months after the start of the accident.

8.15 Containment Shell Transient Pressure and Temperature for the Maximum Credible Accident

The maximum credible accident is a composite accident which combines the more severe features of several depressurization accident sequences. In order to determine the maximum transient pressure and temperature that might result from the MCA, a specific mechanistic sequence was selected that included the elements that would provide the maximum contribution to containment shell pressure and temperature.

The maximum transient pressure resulting from the MCA is shown on Figure 8.15.1. The maximum pressure and temperature conditions of 8.9 psig and 165 F occur

FIG. 8.15.1

approximately 8 min after the accident. The pressure drops to approximately 3 psig and 71 F at four hours after the start of the accident. At this time the pressure would gradually start to rise due to the controlled nitrogen purge, but since venting through the fission product removal system is initiated within 24 hr, the design pressure of the containment shell is not exceeded.

Figure 8.15.2 shows the transient temperature of the containment shell gases and vapors and of the concrete surface following the accident. During the initial period of rising pressure and temperature, the temperature of the gases and vapors exceeds that of the concrete surface. The concrete is acting as an effective heat sink during this period and thus supplements the heat removal capability of the containment shell spray system. Soon after the cooldown of the containment shell contents has started, the vapor and gas temperature is below that of the concrete and the heat stored in the concrete is released. This cross-over occurs at approximately 12 min after the start of the accident.

The pressure and temperature in the reactor building shown in Figures 8.15.1 and 8.15.2 combine the most pessimistic assumptions regarding mass and energy transfer to the reactor building in order to obtain the highest pressure and therefore the greatest driving force for releasing activity to the environment. Immediately following any depressurization accident, one or more of the reactor coolant blowers would be operating to provide adequate core cooling and therefore the emergency cooling system would not be required to cool the core. With the reactor coolant blower(s) operating, nitrogen purge is not required and therefore both pressure and temperature in the reactor building decreases from the values used in the MCA analysis.

FIG. 8.15.2

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TABLE 9.8.1 (Contd.)

MCA Source Term(4)
Activity Release (curies)^a

	Time After Accident	<u>89Sr</u>	<u>90Sr</u>	<u>137Cs</u>	<u>140Ba</u>	<u>140La</u>	<u>131I</u>	<u>133I</u>	<u>135I</u>	<u>85mKr</u>	<u>133Xe</u>	<u>135Xe</u>
9-19	<u>24 hr</u>											
	Initial Activity	374.	6.8	1428.	51.	37.4	4420.	1530.	146.	7.14	10200.	510.
	Oxidation Release	<u>196.</u>	<u>28.1</u>	<u>29.</u>	<u>315.</u>	<u>219.</u>	<u>4284.</u>	<u>4964.</u>	<u>826.</u>	<u>155.</u>	<u>30600.</u>	<u>5440.</u>
	Total	570.	34.9	1460.	366.	256.	8700.	6490.	972.	162.	40800.	5950.
	<u>32 hr</u>											
	Initial Activity	252.	4.6	969.	34.	22.	2679.	796.	43.2	1.4	6630.	189.
	Oxidation Release	<u>182.</u>	<u>26.4</u>	<u>28.</u>	<u>290.</u>	<u>179.</u>	<u>3410.</u>	<u>3536.</u>	<u>336.6</u>	<u>34.3</u>	<u>27472.</u>	<u>2802.</u>
	Total	434.	31.0	997.	324.	201.	6590.	4330.	380.	35.7	34100.	2990.
	<u>40 hr</u>											
	Initial Activity	194.	3.6	748.	25.8	14.8	2040.	469.	14.5	0.3	4896.	80.6
	Oxidation Release	<u>191.</u>	<u>27.9</u>	<u>29.</u>	<u>301.</u>	<u>165.</u>	<u>4012.</u>	<u>2870.</u>	<u>155.7</u>	<u>10.9</u>	<u>27846.</u>	<u>1621.8</u>
	Total	385.	31.5	777.	327.	180.	6050.	3340.	170.	11.2	32700.	1700.
	<u>48 hr</u>											
	Initial Activity	165.	3.1	639.	21.6	11.1	1697.	306.	5.4	0.1	4012.	37.8
	Oxidation Release	<u>216.</u>	<u>31.7</u>	<u>33.</u>	<u>335.</u>	<u>163.</u>	<u>4417.</u>	<u>2482.</u>	<u>76.8</u>	<u>3.6</u>	<u>30192.</u>	<u>1006.4</u>
	Total	381.	34.8	672.	357.	174.	6110.	2790.	82.2	3.7	34200.	1040.
	<u>56 hr</u>											
	Initial Activity	152.	2.8	592.	19.6	8.9	1527.	217.3	2.2	0.02	3536.	19.2
	Oxidation Release	<u>251.</u>	<u>37.1</u>	<u>38.8</u>	<u>384.</u>	<u>166.</u>	<u>5032.</u>	<u>2210.</u>	<u>39.1</u>	<u>1.2</u>	<u>33898.</u>	<u>639.2</u>
	Total	403.	39.9	631.	404.	175.	6560.	2430.	41.3	1.22	37400.	658.

TABLE 9.8.1 (Contd.)

MCA Source Term(4)
Activity Release (curies)^a

Time After Accident	<u>⁸⁹Sr</u>	<u>⁹⁰Sr</u>	<u>¹³⁷Cs</u>	<u>¹⁴⁰Ba</u>	<u>¹⁴⁰La</u>	<u>¹³¹I</u>	<u>¹³³I</u>	<u>¹³⁵I</u>	<u>^{85m}Kr</u>	<u>¹³³Xe</u>	<u>¹³⁵Xe</u>
<u>64 hr</u>											
Initial Activity	146.	2.7	571.	18.6	7.5	1431.	160.8	0.9		3274.	10.2
Oxidation Release	<u>293.</u>	<u>43.5</u>	<u>45.6</u>	<u>445.</u>	<u>167.</u>	<u>5780.</u>	<u>1989.</u>	<u>19.8</u>	<u>0.4</u>	<u>38420.</u>	<u>391.</u>
Total	439.	46.2	617.	464.	175.	7210.	2150.	20.7	0.4	41700.	401.
<u>72 hr</u>											
Initial Activity	143.	2.7	561.	18.	6.4	1370.	121.4	0.4		3091.	5.5
Oxidation Release	<u>338.</u>	<u>50.</u>	<u>52.7</u>	<u>510.</u>	<u>169.</u>	<u>6494.</u>	<u>1799.</u>	<u>10.3</u>	<u>0.1</u>	<u>42500.</u>	<u>252.6</u>
Total	481	52.7	614.	528.	175.	7860.	1920.	10.7	0.1	45600.	258.
<u>81 hr</u>											
Initial Activity	129.	2.4	510.	16.	5.0	1200.	81.3	0.1		2662.	2.5
Oxidation Release	<u>357.</u>	<u>56.1</u>	<u>55.8</u>	<u>530.</u>	<u>153.</u>	<u>6664.</u>	<u>1411.</u>	<u>4.3</u>		<u>42840.</u>	<u>138.</u>
Total	486.	58.5	566.	546.	158.	7860.	1490.	4.4		45500.	141.
<u>96 hr</u>											
Initial Activity	128.	2.4	510.	16.	3.8	1139.	49.3			2455.	0.8
Oxidation Release	<u>442.</u>	<u>69.7</u>	<u>69.7</u>	<u>663.</u>	<u>148.</u>	<u>7922.</u>	<u>1068.</u>	<u>1.1</u>		<u>49300.</u>	<u>56.1</u>
Total	570.	72.1	580.	679.	152.	9060.	1120.	1.1		51800.	56.9
<u>14 days</u>											
Initial Activity	111.	2.4	510.	9.3	0.1	479.				663.	
Oxidation Release	<u>1183.</u>	<u>71.1</u>	<u>214.</u>	<u>1156.</u>	<u>7.3</u>	<u>10200.</u>	<u>1.1</u>			<u>40900.</u>	
Total	1290.	73.5	724.	1165.	7.4	10700.	1.1			41600.	
<u>24 days</u>											
Initial Activity	96.9	2.4	510.	5.4		202.				179.	
Oxidation Release	<u>1326.</u>	<u>71.4</u>	<u>275.</u>	<u>860.</u>		<u>5542.</u>				<u>14140.</u>	
Total	1420.	73.8	785.	865.		5740.				14300.	

TABLE 9.8.1 (Contd.)

MCA Source Term(4)
Activity Release (curies)^a

<u>Time After Accident</u>	<u>89Sr</u>	<u>90Sr</u>	<u>137Cs</u>	<u>140Ba</u>	<u>140La</u>	<u>131I</u>	<u>133I</u>	<u>135I</u>	<u>85mKr</u>	<u>133Xe</u>	<u>135Xe</u>
<u>34 days</u>											
Initial Activity	84.7	2.4	510.	3.1		85.3				48.2	
Oxidation Release	<u>1265.</u>	<u>71.4</u>	<u>301.</u>	<u>547.</u>		<u>2557.</u>				<u>4148.</u>	
Total	1350.	73.8	811.	550.		2640.				4200.	
<u>44 days</u>											
Initial Activity	73.8	2.4	510.	1.8		36.				13.	
Oxidation Release	<u>1146.</u>	<u>71.4</u>	<u>312.</u>	<u>330.</u>		<u>1115.</u>				<u>1159.</u>	
Total	1220.	73.8	822.	332.		1150.				1170.	
<u>54 days</u>											
Initial Activity	64.3	2.4	510.	1.1		15.2				3.5	
Oxidation Release	<u>1013.</u>	<u>71.4</u>	<u>317.</u>	<u>195.</u>		<u>476.</u>				<u>318.</u>	
Total	1080.	73.8	827	196.		491.				322.	
<u>64 days</u>											
Initial Activity	56.1	2.4	510.	0.6		6.4				0.9	
Oxidation Release	<u>891.</u>	<u>71.4</u>	<u>319.</u>	<u>114.</u>		<u>202.</u>				<u>85.</u>	
Total	947.	73.8	829.	115.		208.				86.	

- a. Activity release assumes that 10 fuel element failures can be used to represent the 1000 curies initially in the system; 330 fuel failures occur during the first 30 seconds; total number of failures is 340.

TABLE 9.8.2.1

Doses at the Low Population Boundary
(4600 m)⁽³⁾

<u>Time Interval</u> (hr)	<u>Type of Release</u>	<u>Thyroid Dose</u> (rem)	<u>Bone Dose</u> (rem)	<u>Total Body Dose</u> (rem)
0 - 24	leakage	0.511	0.048	0.006
24 - 81	leakage	0.685	0.121	0.071
24 - 81	venting	<u>0.407</u>	<u>0.015</u>	<u>0.033</u>
Total dose		1.6 rem	0.18 rem	0.11 rem

TABLE 9.8.2.2

Doses at the Exclusion Radius
(1060) for 2-hr Exposure⁽³⁾

<u>Organ</u>	<u>Dose (rem)</u>
Thyroid	0.328
Bone	0.024
Total body	0.004

TABLE 9.8.2.3

Cumulative Doses at the Near Shore
(200 m) of Melton Hill Lake ⁽³⁾

<u>Time (hr)</u>	<u>Thyroid Dose (rem)</u>	<u>Bone Dose (rem)</u>	<u>Total Body Dose (rem)</u>
0.5	1.24	0.08	0.02
1.0	2.53	0.17	0.05
1.5	3.83	0.29	0.07
2.0	5.12	0.38	0.09

b. Dose in Control Room

→ The submersion dose to personnel in the main control room after the MCA is less than 1.5 mrem/hr for periods up to 64 hours following the accident.

→ The internal doses to the thyroid and bone are shown in Table 9.8.2.4.

TABLE 9.8.2.4

Internal Dose to Personnel in Control Room

<u>Time Period</u>	<u>Integrated Dose (rem)</u>	
	<u>Thyroid</u>	<u>Bone</u>
0-24 hr	3.78	1.15
24-72 hr	5.03	2.56
72-81 hr	0.48	0.30
Total	9.29	4.01

The above doses are based on the assumption that there is an infinite cloud present both inside and outside the control room. The concentration of activity in the cloud inside the control room is evaluated for equilibrium conditions with an inleakage into the control room of 5%/day.

The direct radiation dose rate in the main control room from fission products inside the reactor building is less than 1 mrem/hr. In the control room extension, which contains the emergency cooling system controls, the maximum direct radiation dose rate is 20 mrem/hr. This dose rate occurs one day after the accident and then decreases to about 5 mrem/hr after 20 days.

9.8.3 Rainout

The following assumptions were used in the rainout calculation:

- All activity release is at ground level.
- Release rate, including leakage and venting, is chosen as the highest value existing during the MCA.
- Inversion conditions are considered for atmospheric dispersion.
- Particulates and halogens are removed by filter system (Section 6.15.3) with efficiencies of 99 and 95%, respectively. This applies only to portion of source from venting.
- The allowable concentration of unknown mixture of fission products known not to contain radium is $10^{-7} \mu\text{C}/\text{cm}^3$ for public exposure.⁽²⁾

The average concentration of fission products in the Melton Hill Lake, Gallaher Bend area, from 200 to 1060 m from the EGCR was calculated to be $1.6 (\text{MPC})_w$ for an unknown mixture of fission products known not to contain radium.

The maximum direct radiation dose received by a person from rainout on land was calculated as 13.3 mrem/hr at any time during the course of the MCA at any location from 1060 to 10,000 m from the site. The rate of 13.3 mrem/hr occurs at 1060 m.

9.8.4 Dry Fallout

The basic assumptions used in the rainout calculations were also applied to the fallout calculation, with the exception that lapse conditions were used for dry fallout on land. The concentration increase in Melton Hill Lake was calculated based on lapse conditions as $8.9 \times 10^{-9} (\text{MPC})_w$.

The direct radiation dose rate for dry fallout on land does not exceed 1.8 mrem/hr at the maximum dose point.

References:

1. Permissible Dose from External Sources of Ionizing Radiation, National Bureau of Standards Handbook 59, U. S. Department of Commerce, issued September 24, 1954, as amended.
2. Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, National Bureau of Standards Handbook 69, U. S. Department of Commerce (June 5, 1959).
3. Luse, D. F., Hazards Analysis - Radiological Hazards, EGCR Study IV-313, Rev. 1, Allis-Chalmers (July 11, 1963). ←
4. Row, T. H., EGCR Maximum Credible Accident Source Term, ORNL CF 62-7-1 (July 16, 1962).